Status Report – APR1400 (KEPCO E&C/KHNP)
KOREA 2020/05/15

This reactor design is an evolution from the previous **OPR1000**, of which **2** units are operating in [KOREA and https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=KR], of which **8** units are under construction in [KOREA and https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=KR] and UAE and https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=AE].

The reference plant is [SHIN-KORI-3 https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=885] and has a net power output of 1400 MWe.

INTRODUCTION

Indicate which booklet(s): [O] Large WCR [ ] SMR [ ] FR

Korea Electric Power Company (KEPCO) designed the APR1400, an evolutionary reactor which incorporates a variety of engineering improvements to enhance safety, improve economics, and increase reliability of nuclear electricity generation in the Republic of South Korea. This reactor is a 1400 MWe PWR that utilizes innovative active as well as passive safety systems to provide high performance and safe reactor operating conditions. The design evolved from the OPR1000 with higher safety and seismic resistance features. The philosophy that safety and economics go hand-in-hand resulted in a worldwide deployable design that can be tailored to a variety of utility requirements. The company has drawn from its extensive experience in areas of construction, operations and decommissioning in the nuclear industry to incorporate lessons learned from their previous endeavours. A parallel research and design process allowed the incorporation of results from a series of experimental research projects with the purpose of protecting workers, the public and the environment.

Development Milestones

1992	ARR1400 – Start of design
1994	Concept design completed
1999	Engineering design complete
2002	Secure necessary licenses
2008	Start construction of a first full NPP
2016	Commercial operation

Design organization or vendor company (**e-mail contact**):
KEPCO E&C (xylitol@kepco-enc.com, shjun@kepco-enc.com, sleepy@kepco-enc.com)

Links to designer/vendor homepage: www.kepco-enc.com

Detailed Design Description:

Most Recent Licensing Application Support Document, e.g.:

- Design Certification Document (DSD) and Final Safety Evaluation Report (FSAR)
- https://www.nrc.gov/reactors/new-reactors/design-cert/apr1400.html

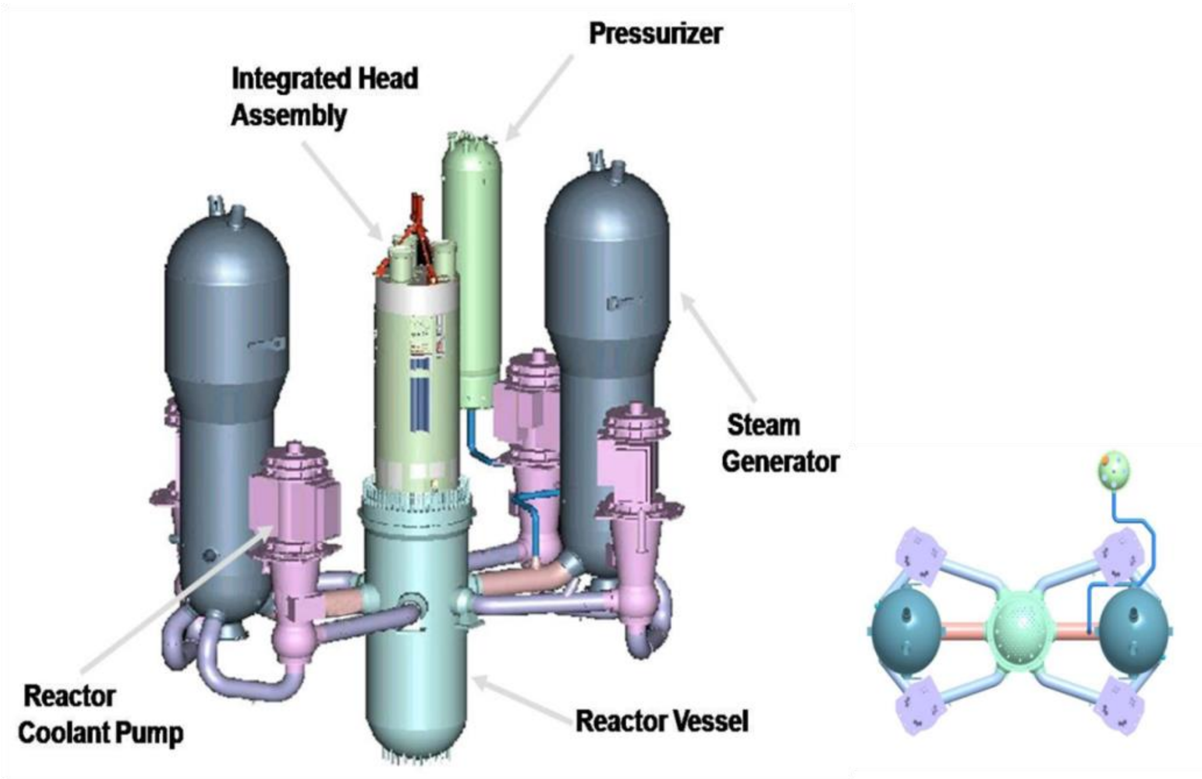


Figure 1. APR1400 Reactor Coolant System Configuration

Table 1: ARIS Category Fields (see also Spreadsheet "Categories") for Booklet

ARIS Category	Input	Select from
Current/Intended Purpose	Commercial – Electric	Commercial – Electric/Non-electric, Prototype/FOAK, Demonstration, Experimental
Main Intended Application (once commercial)	Baseload	Baseload, Dispatchable, Off-grid/Remote, Mobile/Propulsion, Non-electric (specify)
Reference Location	On Coast	On Coast, Inland, Below-Ground, Floating-Fixed, Marine-Mobile, Submerged-Fixed (Other-specify)
Reference Site Design (reactor units per site)	Dual Unit	Single Unit, Dual Unit, Multiple Unit (# units)
Reactor Core Size (1 core)	Large	Small (<1000 MWth), Medium (1000-3000 MWth), Large (>3000 MWth)
Reactor Type	PWR	PWR, BWR, HWR, SCWR, GCR, GFR, SFR, LFR, MSR, ADS
Core Coolant	H_2O	H_2O, D_2O, He, CO_2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Neutron Moderator	H_2O	H_2O, D_2O, Graphite, None, (Other-specify)
NSSS Layout	Loop-type (2 loops)	Loop-type (# loops), Direct-cycle, Semi-integral, Integral, Pool-type
Primary Circulation	Forced (4 pumps)	Forced (# pumps), Natural
Thermodynamic Cycle	Rankine	Rankine, Brayton, Combined-Cycle (direct/indirect)
Secondary Side Fluid	H_2O	H_2O, He, CO_2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Fuel Form	Fuel Assembly /Bundle	Fuel Assembly/Bundle, Coated Sphere, Plate, Prismatic, Contained Liquid, Liquid Fuel/Coolant
Fuel Lattice Shape	Square	Square, Hexagonal, Triangular, Cylindrical, Spherical, Other, n/a
Rods/Pins per Fuel Assembly/Bundle	236	#, n/a
Fuel Material Type	Oxide	Oxide, Nitride, Carbide, Metal, Molten Salt, (Other-specify)
Design Status	Final (with secure suppliers)	Conceptual, Detailed, Final (with secure suppliers)
Licensing Status	Design Licensed (KOREA, USA), Under Construction (8 units), In Operation (2 units)	DCR, GDR, PSAR, FSAR, Design Licensed (in Country), Under Construction (# units), In Operation (# units)

Table 2: ARIS Parameter Fields (see also Spreadsheet "Data") for Booklet

ARIS Parameter	Value	Units or Examples
Plant Infrastructure		
Design Life	60	years
Lifetime Capacity Factor	More than 90	%, defined as Lifetime MWe-yrs delivered / (MWe capacity * Design Life), incl. outages
Major Planned Outages	Every 18 months for refuelling	# days every # months (specify purpose, including refuelling)
Operation / Maintenance Human Resources	11 per rotation (6) / 200 for 2 units	# Staff in Operation / Maintenance Crew during Normal Operation
Reference Site Design	2 Units/Modules	n Units/Modules
Capacity to Electric Grid	1,400 MWe	MWe (net to grid)
Non-electric Capacity	None	e.g. MWth heat at x ºC, m3/day desalinated water, kg/day hydrogen, etc.
In-House Plant Consumption	65 MWe	MWe
Plant Footprint	89,943 m^2	m^2 (rectangular building envelope)
Site Footprint	1,737,973 m^2	m^2 (fenced area)
Emergency Planning Zone	8 km	km (radius)
Releases during Normal Operation	5.31E+2 / 5.92 / 4.22E-2	TBq/yr (Noble Gases / Tritium Gas / Liquids)
Load Following Range and Speed	50 – 100 0.42% per minute	x – 100%, % per minute
Seismic Design (SSE)	0.3g	g (Safe-Shutdown Earthquake)
NSSS Operating Pressure (primary/secondary)	15.5 MPa / 6.9 MPa	MPa(abs), i.e. MPa(g)+0.1, at core/secondary outlets
Primary Coolant Inventory (incl. pressurizer)	339,420	kg
Nominal Coolant Flow Rate (primary/secondary)	21,000 kg/s / 1,134 kg/s	kg/s
Core Inlet / Outlet Coolant Temperature	290.6 ºC / 323.9 ºC	ºC / ºC
Available Temperature as Process Heat Source	307.2 ºC (Tavg)	ºC
NSSS Largest Component	SG	e.g. RPV (empty), SG, Core Module (empty/fuelled), etc.
- dimensions	23m / 6.8m / 781,004kg (dry)	m (length) / m (diameter) / kg (transport weight)
Reactor Vessel Material	SA508	e.g. SS304, SS316, SA508, 800H, Hastelloy N
Steam Generator Design	Vertical, U-tube	e.g. Vertical/Horizontal, U-Tube/ Straight/Helical, cross/counter flow

ARIS Parameter	Value	Units or Examples
Secondary Coolant Inventory	98,910	kg
Pressurizer Design	Separate vessel	e.g. separate vessel, integral, steam or gas pressurized, etc.
Pressurizer Volume	68 m^3 / 33.2 m^3	m^3 / m^3 (total / liquid)
Containment Type and Total Volume	Dry (single) / 9,000 m^3	Dry (single/double), Dry/Wet Well, Inerted, etc. / m^3
Spent Fuel Pool Capacity and Total Volume	20 / 1,740 m^3	years of full-power operation / m^3
Fuel/Core		
Single Core Thermal Power	3,983 MWth	MWth
Refuelling Cycle	18 months	months or "continuous"
Fuel Material	UO_2	e.g. UO_2, MOX, UF_4, UCO
Enrichment (avg./max.)	2.66 % / 4.65 %	%
Average Neutron Energy	1.5eV	eV
Fuel Cladding Material	Zirlo	e.g. Zr-4, SS, TRISO, E-110, none
Number of Fuel "Units"	241 fuel assemblies	specify as Assembly, Bundle, Plate, Sphere, or n/a
Weight of one Fuel Unit	642.4 kg	kg
Total Fissile Loading (initial)	103,300 kg (UO_2)	kg fissile material (specify isotopic and chemical composition)
% of fuel outside core during normal operation		applicable to online refuelling and molten salt reactors
Fraction of fresh-fuel fissile material used up at discharge	86%	%
Core Discharge Burnup	46.5 MWd/kgU	MWd/kgHM (heavy metal, eg U, Pu, Th)
Pin Burnup (max.)	60 MWd/kgU	MWd/kgHM
Breeding Ratio	0.56 ~ 0.60	Fraction of fissile material bred in-situ over one fuel cycle or at equilibrium core
Reprocessing	None	e.g. None, Batch, Continuous (FP polishing/actinide removal), etc.
Main Reactivity Control	Rods, Boron Solution	e.g. Rods, Boron Solution, Fuel Load, Temperature, Flow Rate, Reflectors
Solid Burnable Absorber	Gd_2O_3	e.g. Gd_2O_3,
Core Volume (active)	39.63 m^3	m^3 (used to calculate power density)
Fast Neutron Flux at Core Pressure Boundary		N/m^2-s
Max. Fast Neutron Flux	3.6 X 10^{14} N/m^2-s	N/m^2-s

ARIS Parameter	Value	Units or Examples
Safety Systems		
Number of Safety Trains	Active / Passive	% capacity of each train to fulfil safety function
- reactor shutdown	93 CEAs /	100% /
- core injection	4 (SIS) / 4 (SIT)	50% / 25%
- decay heat removal	2 /	100% /
- containment isolation and cooling	2 /	100% /
- emergency AC supply (e.g. diesels)	2 /	100% /
DC Power Capacity (e.g. batteries)	2,800AH/2h 4,200AH/8h	hours
Events in which ***Immediate Operator Action*** is required	None	e.g. any internal/external initiating events, none
Limiting (shortest) ***Subsequent Operator Action*** Time	0.5 hr / N/A	hours (that are assumed when following EOPs)
Severe Accident Core Provisions	POSRV, Injection to 1'st,&2'nd, IVR	e.g. no core melt, IVMR, Core Catcher, Core Dump Tank, MCCI
Core Damage Frequency (CDF)	$< 10^{-5}$/RY	x / reactor-year (based on reference site and location)
Severe Accident Containment Provisions	Ignitors, ECSBS, Cavity Flooding Sys./PARs	e.g. H_2 ignitors, PARs, filtered venting, etc.
Large Release Frequency (LRF)	$< 10^{-6}$/RY	x / reactor-year (based on reference site and location)
Overall Build Project Costs Estimate or Range (excluding Licensing, based on the Reference Design Site and Location)		
Construction Time (n^{th} of a kind)		months from first concrete to criticality
Design, Project Mgmt. and Procurement Effort		person-years (PY) [DP&P]
Construction and Commissioning Effort		PY [C&C]
Material and Equipment Overnight Capital Cost		Million US$(2015) [M&E], if built in USA
Cost Breakdown	%[C&C] / %[M&E]	
- Site Development before first concrete	/	(e.g. 25 / 10)
- Nuclear Island (NSSS)	/	(30 / 40)
- Conventional Island (Turbine and Cooling)	/	(20 / 25)
- Balance of Plant (BOP)	/	(20 / 10)
- Commissioning and First Fuel Loading	/	(5 / 15) (-----------) (to add up to 100 / 100)
Factory / On-Site split in [C&C] effort	/	% / % of total [C&C] effort in PY (e.g. 60 / 40)

1. Plant Layout, Site Environment and Grid Integration

SUMMARY FOR BOOKLET

The general arrangement of APR1400 is designed based on the twin-unit concept and slide-along arrangement with common facilities such as the compound building which includes radwaste building and access control building. The auxiliary building which accommodates the safety systems and components surrounds the containment building. APR1400 is a pressurized water reactor, and the reactor building essentially coincides with the containment building. The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, pressurizer, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The turbine building houses the turbine generator, the condenser systems, the preheater system, the condensate and feedwater systems, and other systems associated with power generation.

1.1. Buildings and structures, including plot plan

The general arrangement of APR1400 was designed based on the twin-unit concept and slide-along arrangement with common facilities such as the compound building which includes radwaste building and access control building. The general arrangement of the buildings is schematically depicted in Figure 2. The auxiliary building which accommodates the safety systems and components surrounds the containment building. The auxiliary and containment buildings will be built on a common basemat. The common basemat will improve the resistance against seismic events and reduce the number of walls between buildings so that rebar and formwork cost can be reduced.

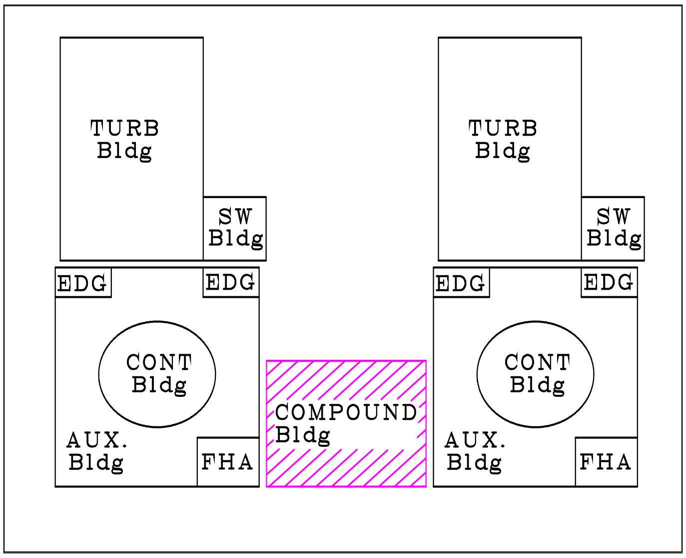


Figure 2. Plant General Arrangement

The layout is highly influenced by safety considerations, in particular, by the physical separation of equipment for the safety systems. The safety injection pumps are located in the auxiliary building in the four quadrants, one pump in each quadrant. This arrangement ensures the physical separation of the pumps, minimizing the propagation of damage due to fire, sabotage, and internal flooding. The emergency diesel generator rooms are also separated and located at the symmetrically opposite sides.

The building arrangement is also designed for the convenience of maintenance, considering accessibility and replacement of equipment. The internal layout of the containment, in particular, is designed to allow the one-piece removal of the steam generator. With proper shielding and arrangement of maintenance space, and careful routing of ventilation air flow, the occupational radiation exposure is expected to be lower than 1 man-Sievert a year.

The design strength of the buildings in the safety category, which are the containment and the auxiliary buildings, is sufficient to withstand the effects of earthquakes up to the Safe Shutdown Earthquake (SSE) of 0.3 g.

1.2.Reactor building

The reactor building is the central building of the plant. APR1400 is a pressurized water reactor, and the reactor building essentially coincides with the containment building. Figure 3 shows a cross-sectional view of the reactor building including a part of the auxiliary building in the vertical direction with the arrangement of major equipment.

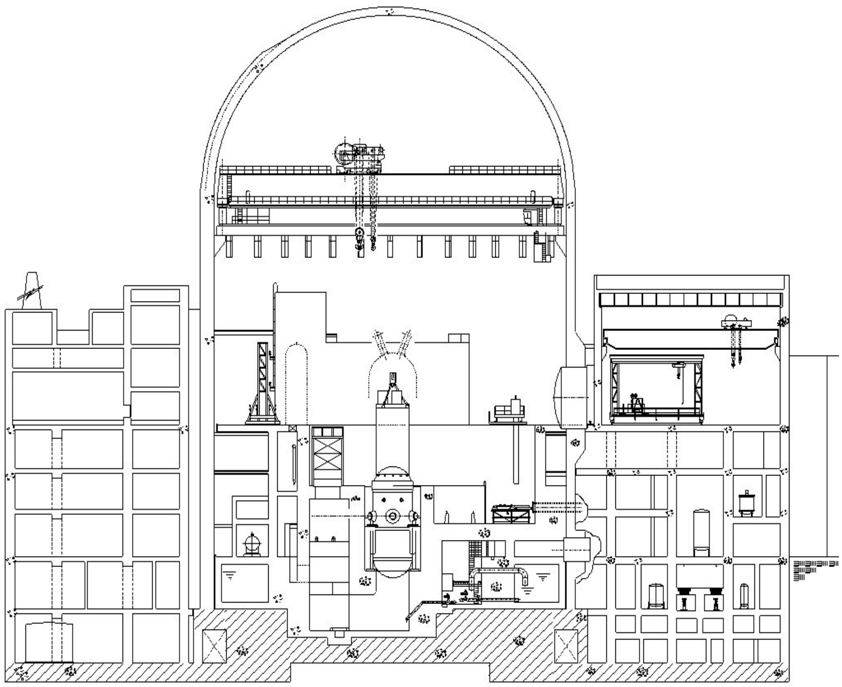


Figure 3. Cross-sectional view of the reactor building

1.3.Containment

The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, pressurizer, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The containment building is designed to sustain all internal and external loading conditions which are reasonably expected to occur during the life of the plant. The containment building is on a common basemat which forms a monolithic structure with the auxiliary building.

The interior arrangement of the containment building is designed to meet the requirements for all anticipated conditions of operations and maintenance, including new and spent fuel handling. There are four main floor levels in the containment: the lowest floor level, called the basement, the highest floor elevation, called the operation floor, and two (2) mezzanine floors in between the basement and operating floors. The two mezzanine floors are designed primarily of steel-supported grating.

The equipment hatch is at the operating floor level, and has an inside diameter of 7.8m (26 ft). This hatch size is selected to accommodate the one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators. Personnel access to the containment is through two hatches, one located at the operating floor level and the other at the plant ground elevation.

The containment is a post-tensioned concrete cylinder with an internal diameter of 45.7m (150 ft) and a hemispherical top dome. There is no structural connection between the free standing portion of the containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections.

1.4.Turbine building

The turbine building houses the turbine generator, the condenser systems, the preheater system, the condensate and feedwater systems, and other systems associated with power generation. The turbine building configuration is simplified for constructability, and the maintainability of the systems is improved by centralizing the condensate polishing system, separating the switchgear building, and rearranging the equipment hatches. There are four main floor levels referred to as the basement, ground level, operating level, and deaerator level.

The turbine building is classified as non-safety related. It has no major structural interface with other buildings except for a seismic interface with the connecting auxiliary building. It is designed such that under SSE conditions, its failure will not cause the failure of safety related structures. The turbine building is located such that the containment building is at the high pressure turbine side on the projection of the turbine shaft. This allows the optimization of the piping and cable routes to the nuclear island. This arrangement also minimizes the risk of damage to safety-related equipment by missiles from the turbine or the generator, in the event of an accident. The vibration problem which occurs during transient loading was minimized by moving the fresh water tank of the steam generator blowdown system to the auxiliary building.

In the APR1400 plant, the 52 inches Last Stage Blade (LSB) of the LP turbine was taken into consideration for the building design. Other items reflected in the general arrangement design are as follows:

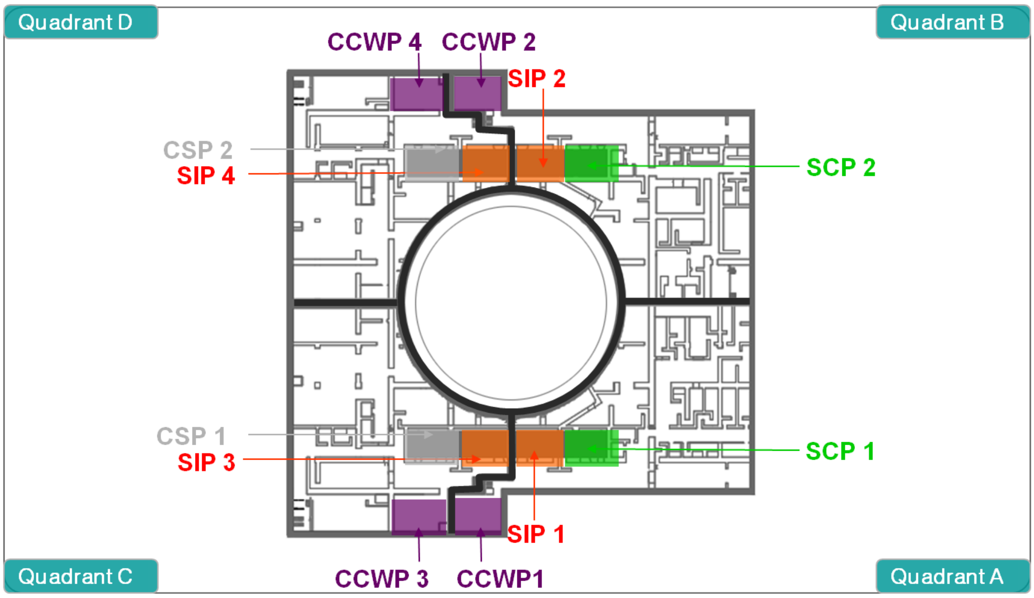
- Relocation of the TBCCW heat exchanger into the turbine building ;
- Relocation of the secondary sample room & lab to the compound building;
- Simplification of contour of the turbine building super structure.

1.5.Other buildings

1.5.1. Auxiliary building arrangement

The auxiliary building completely surrounds the containment building and is on the common basemat which forms a monolithic structure with the containment building. The diesel generator room is built into the auxiliary building. To assure the safety and reliability, the auxiliary building is designed to enhance physical separation for the mitigation of internal flooding and fire propagation as shown in Figure 4.

The auxiliary building houses pumps and heat exchangers for the safety injection system and shutdown cooling system. Also, the auxiliary feedwater tanks and main control room are located in the auxiliary building. For the convenience of operation and maintenance, there is a staging service area in the auxiliary building for installation work in front of the equipment hatch of the containment.



SIP (Safety Injection Pump) SCP (Shutdown Cooling Pump) CSP (Containment Spray Pump) CCWP (Component Cooling Water Pump)

Figure 4. Quadrant arrangement of auxiliary building

The Emergency Diesel Generator (EDG) area is located in the auxiliary building at the ground level. The fuel storage tanks are located on each side of the auxiliary building. The EDGs are arranged as separate entities with dedicated auxiliaries including air supply, exhausts, and cooling systems, so that they are independent of each other in all respects. The EDG areas are arranged to provide routine maintenance facilities and maintenance access space such that work on one EDG will not affect the operability of the other EDG.

1.5.2. Compound building arrangement

The compound building is an integrated building of the radwaste and access control buildings. The compound building consists of an access control facility, a radwaste treatment facility, a hot machine shop, and sampling facilities & lab. The compound building is designed to be shared between two units and is classified as non-safety related.

Radiation shielding is provided wherever required. For the building arrangement design, the protection against natural phenomena and the accommodation of associated environmental conditions were reflected to retain the spillage of potentially contaminated solids or liquids within the building. It has no major structural interface with other building, the access control is made at the ground floor in the compound building.

1.5.3. Switchgear building

The switchgear building is located in the vicinity of the turbine building and all the electrical switchgears are centralized in this area for the convenience of maintenance and efficiency of space allocation.

2. *Technical NSSS/Power Conversion System Design*

SUMMARY FOR BOOKLET

The APR1400 is a Pressurized Water Reactor (PWR) with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated piping are the major components of the RCS. Two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps, are connected to the reactor vessel. One pressurizer is connected to one of the reactor vessel hot legs. All RCS components are located inside the reactor containment building.

The reactor core of APR1400 is designed to generate 3,983 MW thermal power with an average volumetric power density of 100.5 W/cm^3. The reactor core consists of 241 fuel assemblies made of fuel rods containing Uranium Dioxide (UO_2) fuel. The core is designed for an operating cycle of above 18 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of more than 10% to enhance safety and operational performance.

The fuel handling system is designed for a safe and rapid handling and storage of fuel assemblies from the receipt of fresh fuel to the shipment of spent fuel. The major components of the system comprises the refueling machine, the Control Element Assemblie (CEA) change platform, the fuel transfer system, the spent fuel handling machine, the CEA elevator, and the new fuel elevator.

The net effect of the prompt inherent nuclear feedback characteristics, such as fuel temperature coefficient, moderator temperature coefficient, moderator void coefficient, and moderator pressure coefficient, tends to compensate for a rapid increase in reactivity in power operating range. The values of each coefficient of reactivity are consistent with the design basis for net reactivity feedback and analyses that predict acceptable consequences of postulated accidents and AOOs, where such analyses include the response of the reactor protection system (RPS).

The turbine generator plant consists of the main steam, steam extraction, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization was made considering system operability, reliability, availability and economy. The condensate and feedwater systems are designed to deliver the condensate water from the main condenser to the steam generator. The condensate pumps consist of three 50% capacity motor-driven pumps (two operating and one standby). The feedwater pump configuration is selected to be three 50% capacity turbine driven pumps because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation even in the case that one of the feedwater pumps is lost.

The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, pressurizer, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The containment building is designed to sustain all internal and external loading conditions which are reasonably expected to occur during the life of the plant. The containment building is on a common basemat which forms a monolithic structure with the auxiliary building.

The main power system of APR1400 consists of the generator, generator circuit breaker, main trans-former, unit auxiliary transformer and stand-by transformer. The generator is connected to a gas-insulated 345 kV switchyard via the main transformer which is made of three single-phase transformer units. Step-down unit auxiliary transformers are connected between the generator and main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid power supply to the plant auxiliary equipment in the event of failure of the main and unit auxiliary transformers.

The APR1400 I&C system is designed with the network-based distributed control architecture. Operator interface functions and control functions for NSSS, BOP and TG are integrated in common design standards and implemented in common digital system for high functionality, easy operation, and cost effective maintenance. Diversity between safety I&C systems and non-safety I&C systems together with hardwired switches are provided for the defense-in-depth against common mode failure of software in the safety I&C systems.

2.1. Primary Circuit

Since the APR1400 has been evolved from OPR1000, the basic configuration of the nuclear steam supply system is same. As shown in Figure 1, APR1400 has two primary coolant loops and each loop has one steam generator and two reactor coolant pumps in one hot leg and two cold legs arrangement. This two loop/four pump configuration of the reactor coolant system is a well proven design concept through highly reliable operation records of OPR1000 plants.

2.1.1. Primary circuit component description and layout

The APR1400 is a Pressurized Water Reactor (PWR) with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated piping are the major components of the RCS. Two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps, are connected to the reactor vessel. One pressurizer is connected to one of the reactor vessel hot legs. All RCS components are located inside the reactor containment building. The schematic and arrangement of RCS is the same as that of the reference design, APR1400, as shown in Figure 5 thru 7.

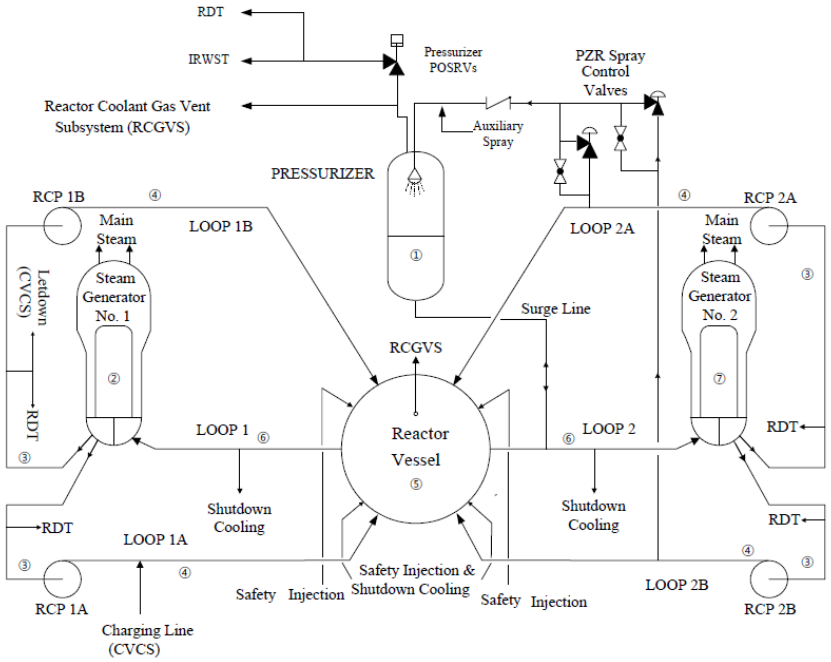


Figure 5 Schematic Flow Diagram of RCS

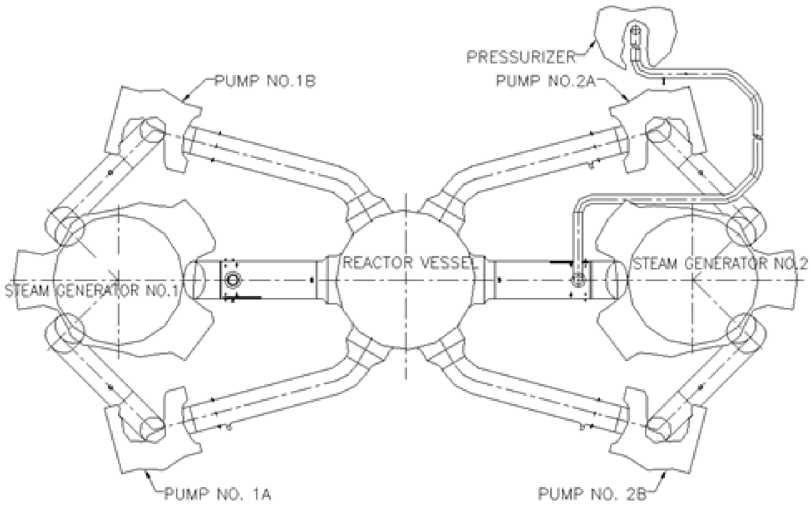


Figure 6 Arrangement Plan View of RCS

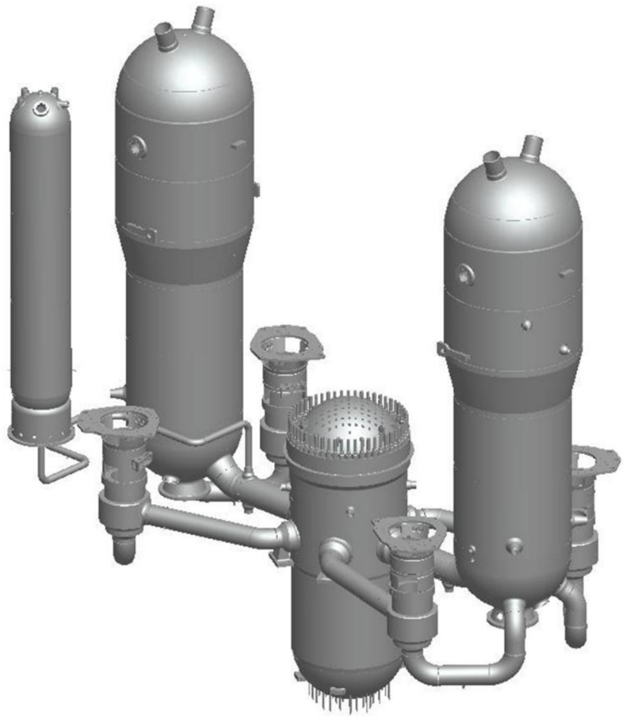


Figure 7 Arrangement Isometric View of RCS

2.1.1.1. Reactor Pressure Vessel

The reactor pressure vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head as shown in Figure 8. The reactor pressure vessel contains internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation components.

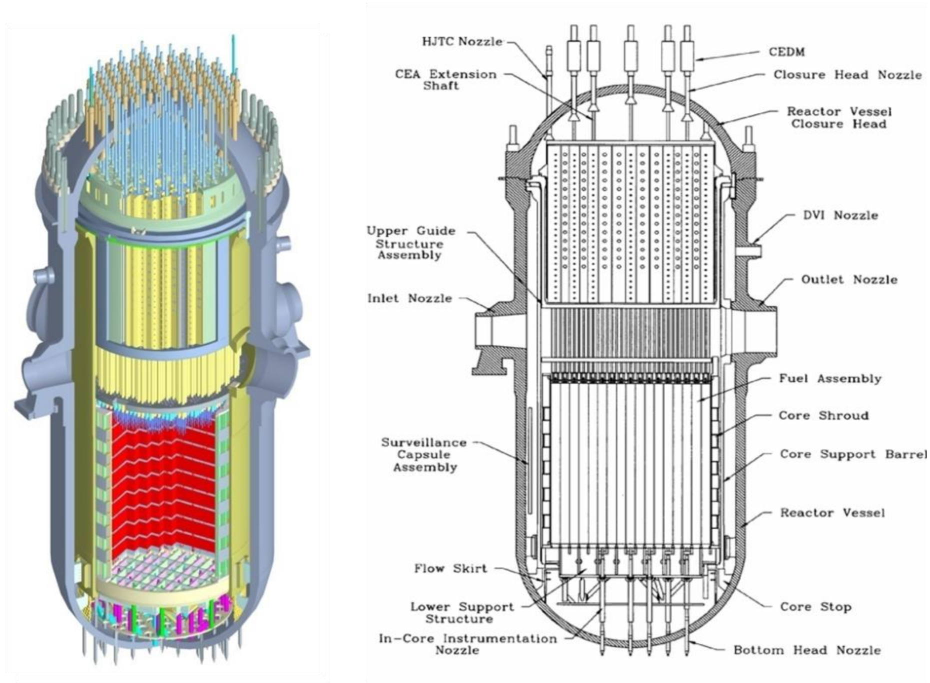


Figure 8. Reactor Vessel and Internals

The structural integrity of the reactor pressure vessel is verified through a structural sizing and fatigue evaluation, which calculates the stresses of the heads, shell and nozzles under thermal and pressure loads.

The Direct Vessel Injection (DVI) nozzles are attached to the reactor vessel for the direct emergency coolant injection as a part of the safety injection system. The location of DVI nozzle is above the cold leg nozzles and determined to avoid the interference with reactor vessel external nozzles and support structure.

The life time of the reactor pressure vessel is extended to 60 years by the use of low carbon steel, which has lower contents of Cu, Ni, P, S as compared to the current design, resulting in an increase of brittle fracture toughness. The inner surface of the reactor vessel is clad with austenitic stainless steel or Ni-Cr-Fe alloy. The reactor vessel is designed to have an end-of-life RT_{NDT} of 21.1℃ (70°F).

The reactor vessel is basically manufactured with a vessel flange, a hemispherical bottom head, and three shell sections of upper, intermediate and lower. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding. Also, four inlet nozzle forgings, two outlet nozzle forgings, four DVI nozzle forgings, and sixty-one ICI nozzles are also welded. The upper closure head is fabricated separately and is joined to the reactor vessel by bolting. The dome and flange are welded together to form the upper closure head, on which the Control Element Drive Mechanism (CEDM) nozzles are welded.

2.1.1.2. Reactor internals

The reactor internals consist of the core support structures, which include the core support barrel, upper guide structure barrel assembly and lower support structure, and the internal structures. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, and to direct the reactor coolant to the core. The primary coolant flows in through the reactor vessel inlet nozzles from the reactor coolant pump, passes through the annulus between the reactor vessel and core support barrel, through the reactor vessel bottom plenum and core, and finally flows out through the outlet nozzles of the reactor vessel connected to the hot legs.

The core support barrel and the upper guide structure are supported at its upper flange from a ledge in the reactor vessel flange. The flange thickness is increased to sustain the enhanced seismic require¬ments. All reactor internals are manufactured of austenitic stainless steel except for the hold-down ring, which is made of high-tension stainless steel. The hold-down ring absorbs vibrations caused by the load to the axial direction of internal structures.

The upper guide structure, which consists of the fuel assembly alignment plate, control element shroud tubes, the upper guide structure base plate, CEA shrouds, and an upper guide structure support barrel, is removed from the core as a single unit during refuelling by means of special lifting rig.

2.1.1.3. Steam Generators

Steam Generator (SG) is a vertical inverse U-tube heat exchanger with an integral economizer, which operates with the RCS coolant in the tube side and secondary coolant in the shell side as shown in Figure 9. The two SGs are designed to transfer the heat of 4,000 MWt from the RCS to the secondary system. The secondary system produces steam to drive the turbine-generator, which generates the net electrical power of 1,400 MWe.

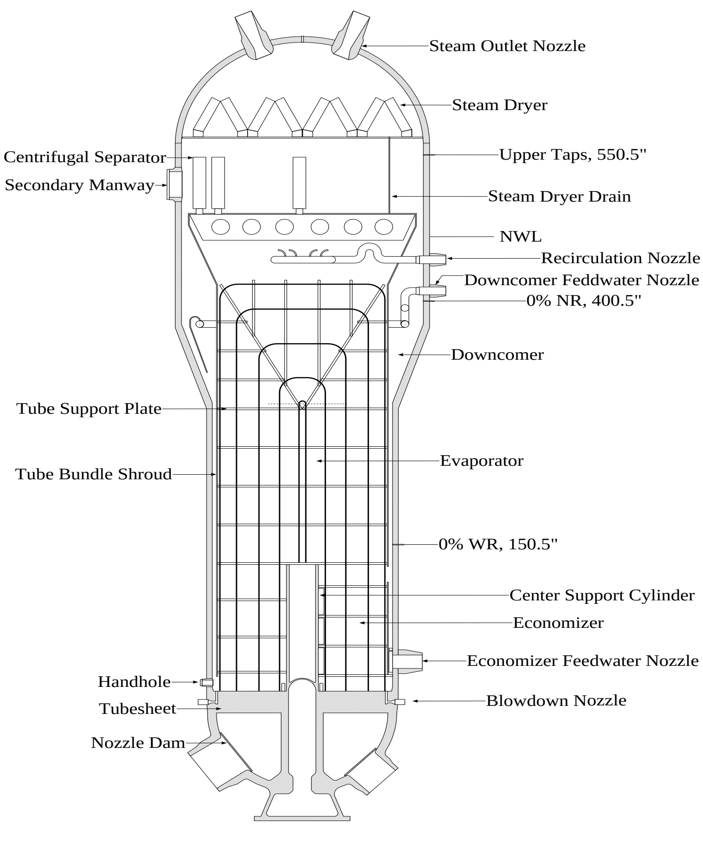


Figure 9. Reactor Vessel and Internals

Moisture separators and steam dryers in the shell side limit the moisture content of the exit steam less than 0.25 w/o during normal full power operation. An integral flow restrictor has been equipped in each SG steam nozzle to restrict the discharge flow in the event of a steam line break. For the maintenance, inspection of equipment condition, and tube-sheet sludge lancing, each SG has a 21 inches man-ways in the cold leg and hot leg side of primary system. Also, two man-ways to allow access to the separator and dryer area of the secondary side, an internal hatch over the top of the tube bundle, two 8 inches hand-holes at the tube-sheet region are also provided.

APR1400 SG uses 13,102 tubes per SG with Alloy 690 as tube material to improve the integrity of SG tube. And the upper tube support bar and plate is designed to prevent SG tube from flow induced vibration. In order to improve the operating margin of the steam generator, the SG tube plugging margin is increased up to 10%. The water volume of SG secondary side is sufficient to provide the dry-out time up to 20 minutes in the event of Total Loss of Feed-Water (TLOFW). This design enhances the capability of alleviating the transients during normal operation by reducing the potential for unplanned reactor trip and the plant safety and the operational flexibility.

To improve the operability, the angle of nozzle in the hot leg side of the primary system is modified to enhance the stability of mid-loop operation. The SG water level control system is designed in such a way that the water level is controlled automatically over the full power operating range.

The economizer feedwater nozzle provides a passage of feedwater to the economizer, which is installed to increase the thermal efficiency of the steam generator at the cold side, and experiences a high temperature gradient. The feedwater nozzles are designed to endure the excessive thermal stress which causes an excessively large fatigue usage factor. The downcomer feedwater nozzle attached in the upper shell of SG also provides small portion of feedwater to the downcomer to facilitate internal recirculation flow. 10% of full power feedwater flow is provided to the downcomer feedwater nozzle and the remaining to the economizer nozzle at a reactor power higher than 15%, below which all feedwater is supplied to the donwncomer nozzle.

2.1.1.4. Pressurizer

The pressurizer is a vertically mounted cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, Pilot Operated Safety Relief Valves (POSRVs), and pressure and level instrumentations. A man-way is provided on the top head for access for inspection of the pressurizer internals. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. The pressurizer maintains RCS pressure and inventory within specified limits in conjunction with the pressurizer pressure and level control systems against all normal and upset operating conditions without reactor trip.

The pressurizer, having a total internal free volume of 68.9m3 (2,400 ft3), is to maintain an operating pressure and temperature of the reactor coolant system. Four(4) POSRVs are adopted instead of two Safety Depressurization System (SDS) valves and three Pressurizer Safety Valve (PSV) of the conventional plant, OPR1000. It provides more reliability in overpressure protection function and more convenience in maintenance activities. The RCS inventory that would discharge through the POSRV under accident conditions is directed to the IRWST and quenched so that the contamination of the containment environment is significantly reduced.

2.1.1.5. Integrated Head Assembly (IHA)

The reactor vessel upper closure head area of the conventional plant consists of CEDM cooling system, cooling shroud assembly, heated junction thermocouples, missile shielding structure, and head lift rig. These components are usually disassembled, separately stored, and reassembled during every refuelling outage. The IHA is a structure to combine and integrate all the reactor vessel closure head area structures into one assembly as shown in Figure 10.
The primary purpose of the IHA is to assemble all the head area structures, components, and cable system and their supports into one assembly so that the refuelling time can be reduced from such operational activities as installation and removal of head area components. Also, the IHA contributes to the reduction of radiation exposures to the maintenance crew since the dissembling and assembling time of the reactor vessel head is reduced.

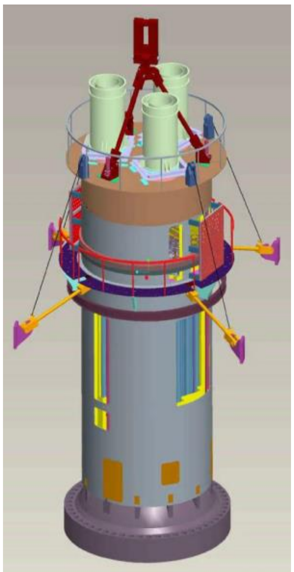


Figure 10. Integrated Head Assembly

2.1.1.6. Reactor Coolant Pumps

The reactor coolant pumps circulate the coolant between the reactor vessel and the steam generators for heat transfer from the reactor core to the SGs. There are two pumps for each coolant loop, located in each cold leg. The pump is a single-stage centrifugal unit of vertical type, driven by a 13,320 hp electric motor. Leaktightness of the shaft is ensured by a mechanical seal designed to prevent leaking against the full internal pressure in the pump.

2.1.1.7. RCS Piping

The Leak-Before-Break (LBB) principle is adopted for the piping system of APR1400, since the pipe whip restraint and the support of the jet impingement shield in the piping system of earlier plants are expensive to build and maintain, and lead to a potential degradation of plant safety. The LBB principle is applied to the main coolant lines, surge lines, and pipes in the shutdown cooling system and the safety injection system. The application of LBB reduces the redundant supports of the pipe in the NSSS pipe system since the dynamic effects of postulated ruptures in the piping system can be eliminated from the design basis. Therefore, the cost of design, construction and maintenance is reduced.

2.1.2. Reactor auxiliary systems

2.1.2.1. Chemical and Volume Control System (CVCS)

The CVCS of APR1400 is not required to perform safety functions such as safe shutdown and accident mitigation. This system is basically for the normal day-to-day operation of the plant. The components related to charging and letdown function, however, are designed as a safety grade and reinforced to assure the reliability for normal and transient conditions. Two centrifugal charging pumps and a flow control valve provide required charging flow. For normal operation, only one charging pump is used to supply the required minimum flow of 12.6 kg/s.

The letdown flow from the reactor coolant system passes through the regenerative and letdown heat exchanger, where an initial temperature reduction takes place. Pressure reduction occurs at the letdown orifice and the letdown control valve. Following temperature and pressure reduction, the flow passes through a purification process at the filters and ion exchangers. After passing through the purification process, the letdown flow is diverted into the Volume Control Tank (VCT) which is designed to provide a reservoir of reactor coolant for the charging pumps and for the dedicated seal injection pumps for the reactor coolant pumps.

2.1.2.2. Component Cooling Water System

The Component Cooling Water System (CCWS) is a closed loop cooling system that, in conjunction with the Essential Service Water System (ESWS) and Ultimate Heat Sink (UHS), removes heat generated from the plant's essential and non-essential components connected to the CCWS. Heat transferred by these components to the CCWS is rejected to the ESWS via the component cooling water heat exchangers. The system is designed to have the cross connection between two divisions to enhance the plant availability and maintenance flexibility.

2.1.2.3. Reactor Coolant Gas Vent System (RCGVS)

The RCGVS is a part of the Safety Depressurization and Vent System (SDVS). The reactor coolant gas vent valves are mounted at the top of the pressurizer and reactor vessel head. The size of the vent line is increased to have sufficient capacity to vent one-half of the RCS volume in one hour assuming a single failure.

2.1.2.4. Steam Generator Blowdown System

The functions of the SG blowdown system are to control SG secondary side water chemistry and to remove sludge from the SG tube support plates. One flash tank can accom¬modate normal and high capacity blowdown flow rates. To remove dynamic loading due to two-phase flow, the flash tank for blowdown is located in the auxiliary building near the containment. Bypass lines to the condensers are installed to overcome unavailability of the flash tank or the processing system.

2.1.2.5. Primary Sampling System

The primary sampling system is designed to collect and deliver representative samples of liquids and gases in various process systems to the sample station for chemical and radiological analysis. The system permits sampling during normal operation, cooldown and post-accident modes without requiring access to the containment. Remote samples can be taken from the fluids in high radiation areas without requiring access to these areas.

2.1.3. Operating modes

APR1400 is designed to be used for various operating modes not only for the base load full power operation but also for a part load operation such as the load following operation. A standard 100-50-100% daily load follow operation has been considered in the reactor core design as well as in the plant control systems.

In addition, various load maneuvering capabilities are considered in the design such as up to 10% step change in load, +/- 5%/min ramp load changes. Also, it has the house load operation capability during a sudden loss of load up to 100% (full load rejection) in which plant control systems automatically control the plant at 3~5% power level without causing any reactor trips or safety system actuations.

In case of turbine generator trip from any power levels including full power, APR1400 prevents reactor trip and maintains reactor power at reduced level using Reactor Power Cutback System (RPCS) and other control systems. This feature shortens outage time to return to power operation after a problem shooting and enhances plant safety by preventing unnecessary reactor trips. Also, APR1400 control system automatically controls plant parameters and prevents reactor trip during a loss of one or two operating main feedwater pumps event occurring at 100% power operation with 3 main feedwater pumps in service.

2.2. Reactor Core and Fuel

The reactor core of APR1400 is designed to generate 3,983 MW thermal power with an average volumetric power density of 100.5 W/cm^3. The reactor core consists of 241 fuel assemblies made of fuel rods containing Uranium Dioxide (UO_2) fuel. The number of Control Element Assemblies (CEAs) used in the core is 93 in which 81 CEAs are full-strength reactivity control assemblies and the rest are part-strength CEAs. The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets, while Inconel alloy 625 is used as the absorber material for the part-strength control rods.

The core is designed for an operating cycle of above 18 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of more than 10% to enhance safety and operational performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber of gadolinium (Gd_2O_3) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by means of 61 fixed In-Core Instrumentation (ICI) assemblies.

Loading mixed oxide (MOX) fuel up to 1/3 core is considered in the core design. Eight additional reserve CEAs is installed to increase the reactivity control capability, if necessary, for MOX fuel loadings. Also, the APR1400 core is designed to be capable of daily load following operation.

The fuel assembly consists of fuel rods, spacer grids, guide tubes, and upper and lower end fittings. 236 locations of each fuel assembly are occupied by the fuel rods containing UO_2 pellets or the burnable absorber rods containing Gd_2O_3-UO_2 in a 16 × 16 array. The remaining locations are 4 CEA guide tubes and 1 in-core instrumentation guide tube for monitoring the neutron flux shape in the core.

The HIPER16^{TM} fuel design has the capability of a batch average discharge burn-up higher than 55,000 MWD/MTU and the HIPER16^{TM} design has increased overpower margin in comparison with the previous fuel design (PLUS7^{TM}). A schematic diagram of fuel assembly is shown in Figure 11. The HIPER16^{TM} mid-grid design has high through-grid dynamic buckling strength for the enhanced seismic performance. The top nozzle has easy reconstitutability features and holddown spring force has optimized to reduce the fuel assembly bow. The guide tube has high seismic load capability and inner dashpot tube to minimize the fuel assembly bow. The debris filtering and capturing features are implemented in the bottom grid by combining the debris filtering bottom grid and the long bottom end plug to reach the target of zero fuel failure. The bottom nozzle has a low pressure drop features with rectangular flow holes.

The integrity of HIPER16^{TM} fuel has been enhanced by increasing the fretting wear resistance and debris filtering efficiency. The optimized holddown spring force will reduce the HIPER16^{TM} fuel assembly bow. The safety of HIPER16^{TM} fuel has been enhanced by increasing the seismic performance which is related to the spacer grid crush strength and dynamic stiffness.

KEPCO NF is a fuel design and fabrication company that has been responsible for the fuel supplied to all nuclear power plants in Korea for decades. Its major activities include initial and reload core design, fuel development, fuel assembly and component manufacture, and fuel services.

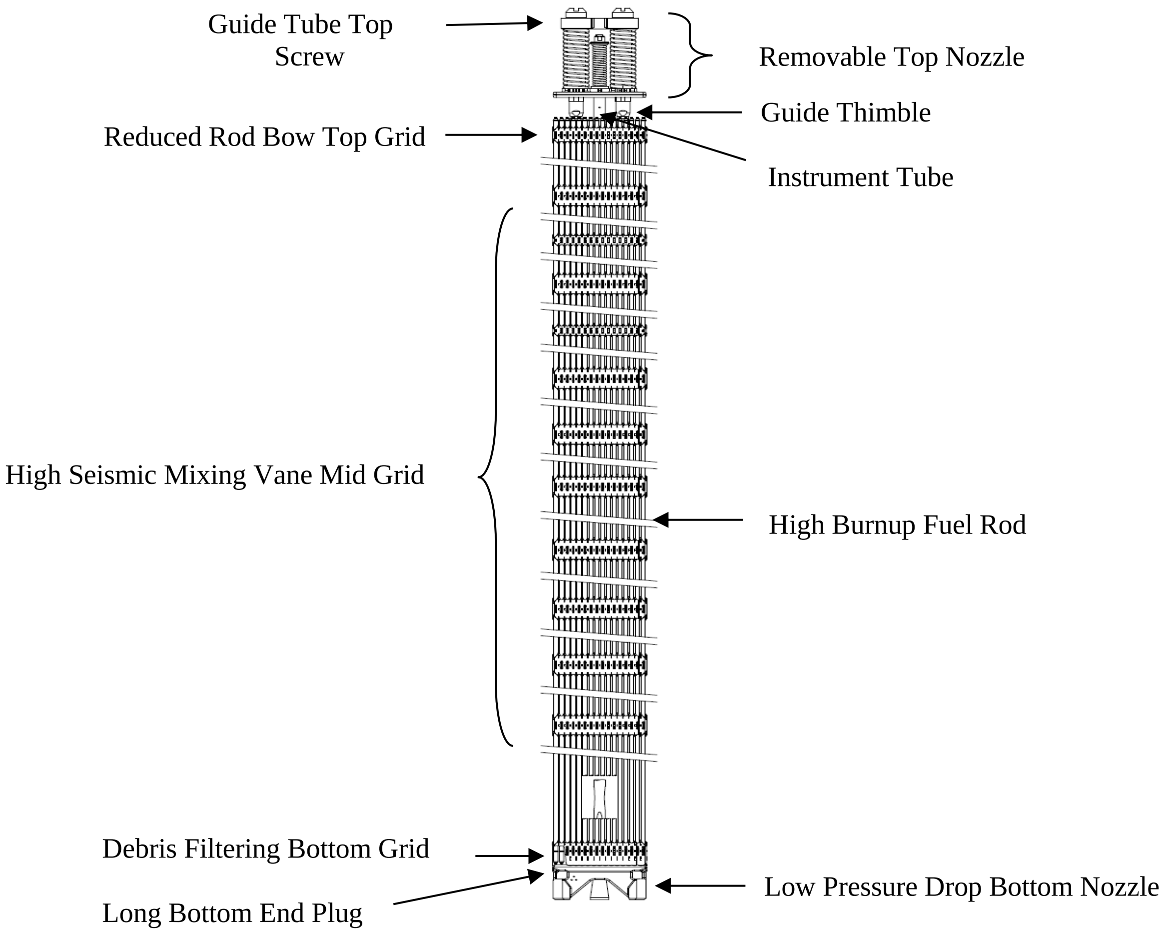


Figure 11. HIPER16™ Fuel Assembly

2.3. Fuel Handling

The fuel handling system is designed for a safe and rapid handling and storage of fuel assemblies from the receipt of fresh fuel to the shipment of spent fuel.

The major components of the system comprises the refuelling machine, the CEA change platform, the fuel transfer system, the spent fuel handling machine, the CEA elevator, and the new fuel elevator. The fuel transfer system moves the fuel between the containment building and the fuel handling area in the auxiliary building through the fuel transfer tube. The refuelling machine is located in the containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The spent fuel handling machine, located in the fuel building, carries fuel to and from the fuel transfer system, the fresh fuel elevator, the spent fuel storage racks, and the spent fuel shipping cask areas.

2.4. Reactor Protection

The net effect of the prompt inherent nuclear feedback characteristics, such as fuel temperature coefficient, moderator temperature coefficient, moderator void coefficient, and moderator pressure coefficient, tends to compensate for a rapid increase in reactivity in power operating range.

The values of each coefficient of reactivity are consistent with the design basis for net reactivity feedback and analyses that predict acceptable consequences of postulated accidents and AOOs, where such analyses include the response of the reactor protection system (RPS).

The core design lifetime and fuel replacement program presented are based on a refueling interval of approximately 18 months with one-third of the fuel assemblies replaced during each refueling outage.

The reactor and the instrumentation and control systems are designed to detect and suppress xenon-induced power distribution oscillations that could, if not suppressed, result in conditions that exceed the specified acceptable fuel design limits. The design of the reactor and associated systems precludes the possibility of power level oscillations.

Power distribution and coolant conditions are controlled so that the peak linear heat rate (LHR) and the minimum departure from nucleate boiling ratio (DNBR) are maintained within operating limits supported by the accident analyses in regard to the correlations between measured quantities, power distribution, and uncertainties in the determination of power distribution. The core operating limit supervisory system (COLSS) indicates to the operator how far the core is from operating limits and provides an audible alarm should an operating limit be exceeded. The COLSS continually assesses the margin of the LHR and DNBR operating limits.

In addition to the COLSS monitoring, the RPS core protection calculators (CPCs) continually monitor the core power distribution and DNBR by processing reactor coolant data, neutron flux signals from ex-core detectors, and input from redundant reed switch assemblies that indicate CEA position. In the event the power distributions or other parameters are perturbed as the result of an AOO that would violate fuel design limits, the high local power density or low DNBR trips in the RPS initiate a reactor trip.

2.5. Secondary Side

2.5.1. Turbine generator description

The turbine generator plant consists of the main steam, steam extraction, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization was made considering system operability, reliability, availability and economy.

The turbine generator system is designed to be capable of operation at 3% house load for a period of at least 4 hours without any detrimental effects in the system, and capable of startup to full load from the cold condition in 8 hours including rotor preheat.

The main steam lines and the high-pressure turbine are designed for a steam pressure of 6.9 MPa (1,000 psia), and two reheater stages are provided between the high pressure and the low pressure turbines. The generator is a three phase, 4-pole unit operating at 1800 rpm.

The capacity, response and modulation capabilities of the turbine bypass system are designed to make the turbine capable of withstanding a 100% generator load rejection without trip of the reactor or the turbine. The total flow capacity of the turbine bypass system is designed to be 55% of the turbine steam flow at full load steam pressure.

2.5.2. Condensate and feed water systems

The condensate and feedwater systems are designed to deliver the condensate water from the main condenser to the steam generator. The condensate pumps consist of three 50% capacity motor-driven pumps (two operating and one standby). The feedwater pump configuration is selected to be three 50% capacity turbine driven pumps because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation even in the case that one of the feedwater pumps is lost.

During the shutdown and startup, a motor-driven startup feedwater pump provides feedwater from the deaerator storage tank or condensate tank. The startup feedwater pump is capable of providing up to 5% of full power feedwater flow to both steam generators. On-line condensate polishers, which can operate in full and partial flow, as well as in bypass mode, are provided to maintain proper water chemistry during normal power operation. In the feedwater systems, the feedwater heaters are installed in 7 stages and arranged horizontally for easy maintenance and high reliability.

2.5.3. Auxiliary systems

2.5.3.1. Turbine bypass system

The turbine bypass system is provided to dissipate heat from the reactor coolant system during the turbine and/or the reactor trip. The OPR1000 and the APR1400 plant have the same capability of relieving 55% of full load main steam flow. In the case of OPR1000, 15% are dumped to the atmosphere and 40% are discharged to the main condenser while the APR1400 plant discharges the total 55% directly into the main condenser.

2.5.3.2. Turbine Building Open Cooling Water System

The Turbine Building Open Cooling Water System (TBOCW) supplies seawater to the service side of the turbine building closed cooling water heat exchangers. The APR1400 plant does not need the TBOCW pump which is installed in the OPR1000 to supply seawater as a heat sink for the plant. In the APR1400 plant design, the TBOCW system interfaces with the Circulating Water (CW) system to take the fresh seawater and discharge the heated seawater to the CW discharge conduit. This design concept reduces the plant capital cost.

2.5.3.3. Condenser vacuum system

The Condenser Vacuum (CV) system supports the plant startup and maintains the condenser vacuum by continuously removing non-condensible gases and air. The system consists of four 33-1/3 % capacity condenser vacuum pumps which are used to draw down the condenser shell pressure. These pumps are also used for

"Holding mode" during normal operation without the steam jet air ejectors. In addition, the radiation level in the CV system discharge is continuously displayed on the radiation monitoring system in the main control room. The APR1400 plant is designed to combine the system discharge and the deaerator normal vent flow line to reduce the number of radiation monitors.

2.6. Containment/Confinement

The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, pressurizer, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The containment building is designed to sustain all internal and external loading conditions which are reasonably expected to occur during the life of the plant. The containment building is on a common basemat which forms a monolithic structure with the auxiliary building.

The interior arrangement of the containment building is designed to meet the requirements for all anticipated conditions of operations and maintenance, including new and spent fuel handling. There are four main floor levels in the containment: the lowest floor level, called the basement, the highest floor elevation, called the operation floor, and two (2) mezzanine floors in between the basement and operating floors. The two mezzanine floors are designed primarily of steel-supported grating.

The equipment hatch is at the operating floor level, and has an inside diameter of 7.8m (26 ft). This hatch size is selected to accommodate the one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators. Personnel access to the containment is through two hatches, one located at the operating floor level and the other at the plant ground elevation.

The containment is a post-tensioned concrete cylinder with an internal diameter of 45.7m (150 ft) and a hemispherical top dome. There is no structural connection between the free standing portion of the containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections.

Containment design and parameters performance characteristics are as follows.

Design Parameter	Design value
Containment internal design pressure, kg/cm^{2}G (psig)	4.22 (60)
Containment design temperature, °C (°F)	143.3 (290.0)
Containment external design pressure, kg/cm^{2}G (psig)	0.28 (4.0)
Containment Net free volume, m^3 (ft^3)	8.8576×10^4 (3.128×10^6)
Design leak rate, First 24 hours (% free volume/day)	0.1
Design leak rate, After 1 day (% free volume/day)	0.05

The containment systems include the containment spray system, containment air purification and clean-up systems, containment isolation system, and containment combustible gas control system.

The containment spray system (CSS) is a safety-related system which removes heat and reduces the concentration of radionuclides released from the containment atmosphere and transfers the heat to the component cooling water system following events which increase containment temperature and pressure.

The containment isolation system (CIS) provides a safety-related means of isolating fluid systems that passes through the containment penetrations to confine the release of any radioactivity from the containment following a postulated DBA. The CIS provides a pressure barrier at each of these containment penetrations.

The containment hydrogen control system is designed to control combustible gas, primarily hydrogen gas (H_2), inside the containment within the acceptable limits. Combustible gas is controlled by passive autocatalytic recombiners or hydrogen igniters with consideration of hydrogen generation during a severe accident.

2.7. Electrical, I&C and Human Interface

2.7.1. Electrical systems

The main power system of APR1400 consists of the generator, generator circuit breaker, main trans-former, unit auxiliary transformer and stand-by transformer. The generator is connected to a gas-insulated 345 kV switchyard via the main transformer which is made of three single-phase transformer units. Step-down unit auxiliary transformers are connected between the generator and main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid power supply to the plant auxiliary equipment in the event of failure of the main and unit auxiliary transformers.

The following table describes the turbine and generator design data.

Turbine Generator Design Data	
Turbine type	Tandem-compound
Operating speed	1,800 rpm
Generator output	1,425 MWe at 0.090 kg/cm^{2}A
Voltage	24 kV nominal
Frequency	50 or 60 Hz, three phase

The normal power source for non-safety and permanent non-safety loads is the off-site power source and the generator. If the normal power source is not available, the permanent non-safety loads are covered by two alternative sources: one from the stand-by off-site power source (via the stand-by transformer) and the other from one non-1E alternate AC power source.

The electrical one line diagram of APR1400 is shown in Figure 12.

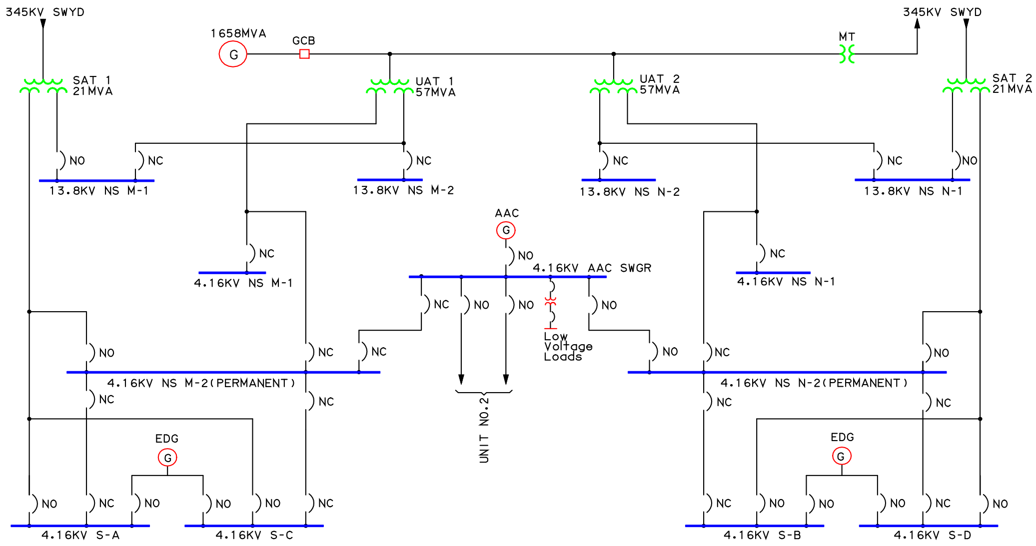


Figure 12. One line diagram

The main features of the electrical system configuration are:

- Two independent off-site power sources of 345 kV
- One main transformer consisting of three single-phase step-down transformers, and two three-winding unit auxiliary transformers for power delivery and supply during normal operation mode
- Two Class 1E emergency Diesel Generators (DGs) to provide on-site stand-by power for the Class 1E loads
- An Alternate AC (AAC) source to provide power for equipment necessary to cope with station blackout at least for 8 hours. For the diversity of emergency electrical power sources, the gas turbine type is selected for AAC
- Automatic transfer of power source from unit auxiliary transformers to standby auxiliary transformers in the event of loss of power supply through the unit auxiliary transformers
- Four independent Class 1E 125V DC systems for each RPS channel
- Two non-class 1E 125V DC systems and one non-class 1E 250V DC system
- AC voltage levels of 13.8 kV and 4.16kV for medium, 480 V and 120V for low voltages

The electric power necessary for the safety-related systems is supplied through 4 alternative ways: firstly, the normal power source, i.e., the normal off-site power and the in-house generation; secondly, the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer; thirdly, the on-site standby power supply, i.e., two diesel generators; and finally, the alternative AC source.

Among these power sources, the on-site standby power is the most crucial for safety; it should be available in any situation. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the equipment to ensure reliability and redundancy of power sources.

The on-site power supply is ensured by two independent Class 1E diesel generator sets; each of them is located in a separated building and is connected to one 4.16 kV safety bus.

The alternate AC source adds more redundancy to the electric power supply even though it is not a safety grade system. The non-class 1E alternate AC is provided to cope with Loss-of-Off-site-Power (LOOP) and Station Blackout (SBO) situation which have a high potential of transients leading to severe accidents. The alternate AC source is sized with sufficient capacity to accommodate the loads on the safety and the permanent non-safety buses

2.7.2. I&C design architecture and applicable standards

APR1400 is, like most of the advanced reactors being developed world-wide, equipped with digitized I&C systems and computer-based control room Man-Machine Interface (MMI), reflecting the status of modern electronics and computer technologies. The I&C and control room concept implemented in the APR1400 design is schematically depicted in Figure 13.

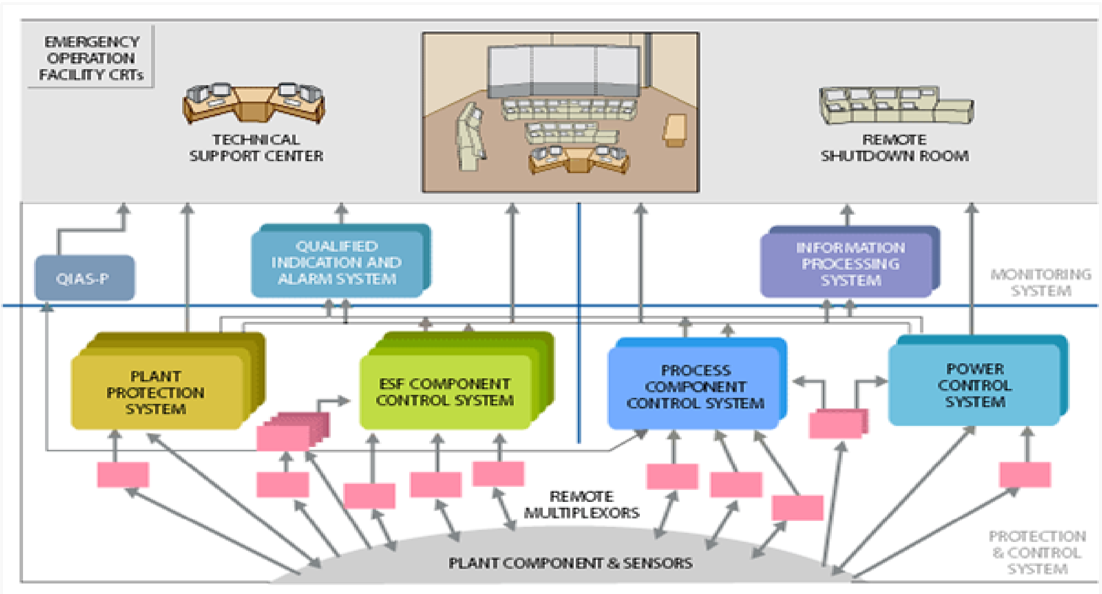


Figure 13. MMIS configuration

The APR1400 I&C system is designed with the network-based distributed control architecture. In this architecture, operator interface functions and control functions for NSSS, BOP and TG are integrated in common design standards and implemented in common digital system for high functionality, easy operation, and cost effective maintenance. Diversity between safety I&C systems and non-safety I&C systems together with hardwired switches are provided for the defense-in-depth against common mode failure of software in the safety I&C systems.

The main features of the I&C system are the use of Distributed Control System (DCS) and microprocessor-based Programmable Logic Controllers (PLCs) for the control and protection systems, and the use of UNIX workstations and industrial PCs for data processing systems.

To protect against common mode failures in software due to the use of software-based I&C systems, DCS and PLCs will be required in the redundant systems for diversity. For data communication, a high-speed fiber optic network based on standard protocols is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission.

Human factor engineering is an essential element of the Main Control Room (MCR) design and the human factor engineering principles are systematically employed to ensure safe and error-free operation. For the successful completion of the APR1400 MMI design process, a multidisciplinary team of human factor specialists, computer specialists, system engineers, and plant operators worked together as a team from the stage of conceptual design through the validation process.

2.7.2.1. Reactor protection and other safety systems

The Plant Protection System (PPS) includes the electrical, electronic, networking, and mechanical devices to perform the protective functions via the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS). The RPS is the portion of the PPS that acts to trip the reactor when the monitored conditions approach specified safety settings and the ESFAS activates the engineered safety systems by safety injection actuation signal and the auxiliary feedwater actuation signal, and etc.

The reactor protection system and other safety-related systems are designed to use the off-the-shelf digital equipment which is commercially available to standardize the components and minimize the maintenance cost with the consideration of diversity. A high degree of conservatism is required in the design of the safety-related systems, and therefore, design principles such as redundancy, diversity, and segmentation have been incorporated in order to achieve both the desired availability and reliability of these systems.

A high reliability of the protection system is ensured by self-diagnostics, and automatic functional tests through surveillance using four independent channels. The redundant and fault tolerant configuration on controllers and the use of fiber-optics to isolate communications will increase system availability and maintainability.

A detailed software development program for software-based Class 1E systems were produced and applied as a guideline to ensure completeness of the software implementation, verification and validation process. Several critical safety systems were evaluated through prototyping and design verification programs.

2.7.2.2. Control Room Facility

The MCR provides operator consoles, safety console, large display panel (LDP), auxiliary panel, and other equipment necessary for the safe and reliable operation of the plant.

Each operator console contains information flat panel displays (IFPD), pointing devices, and ESF-CCS soft control modules (ESCM). The safety console provides the operator with credited backup control, alarm, and indication. The MCR operator consoles and safety console are designed to maintain structural integrity, such that no control room missile hazards result as a consequence of a seismic event.

The remote shutdown room (RSR) is provided to allow emergency shutdown from outside the MCR.

The layout of the MCR is shown in Figure 14.

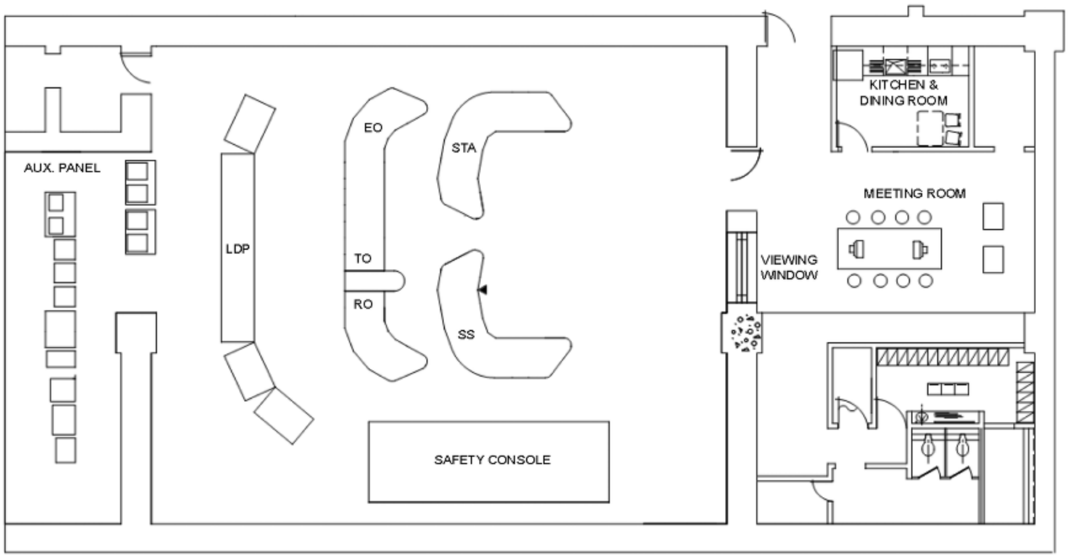


Figure 14. Layout of Main Control Room

3. *Technology Maturity/Readiness*

SUMMARY FOR BOOKLET

In republic of Korea, Shin-Kori units 3&4 (SKN 3&4), the first twin APR1400, construction project was launched with the major supply and construction contracts signed in August 2006. The Operating License (OL) for Unit 3 was issued on October 30, 2015, and on February 1, 2019 for Unit 4. Through successful commissioning and startup tests, commercial operation was started on December 20, 2016 for Unit 3, and August 29, 2019 for Unit 4. The second project for the construction of Shin-Hanul units 1&2(SHN 1&2) is in its final stages with the goal of commercial operation in October 2020 for Unit 1 and August 2021 for Unit 2. The third project for the construction of Shin-Kori units 5&6(SKN 5&6) is in progress with the target commercial operation in March 2023 for Unit 5 and in June 2024 for Unit 6. The design of SHN 1&2 and SKN 5&6 is basically the same as SKN 3&4 except that the MMIS and the Reactor Coolant Pumps (RCP) are manufactured and supplied by domestic vendors. The MMIS and RCP are the results of years of R&D effort funded by government.

On December 27, 2009, the United Arab Emirates (UAE) and Korea signed a historical prime contract for supplying 4 units of APR1400 to the UAE, the first nuclear power plant in the Arab countries. The Preliminary Safety Analysis Report (PSAR) of Barahah Nuclear Power Plant (BNPP) Units 1&2 was submitted to the Federal Authority Nuclear Regulation (FANR) in December 2010 and the Construction License (CL) was issued in July 2012. The construction of Unit 1 begun in July 2012 following the first of nuclear safety concrete, while Unit 2 begun in May 2013. The FSAR of BNPP Units 1&2 was submitted to the FANR in March 2015 and the unit 1 has received operation license in February 2020.

The APR1400 has obtained design certification from U.S. Nuclear Regulatory Commission (NRC), and also won the certification from European Utility Requirements (EUR) organization. In 2017, EUR organization certified the APR1400 design for its compliance with the EUR. In 2019, the U.S. NRC issued final Design Certification (DC) for the APR1400 design. The acquisition of DC is significant in that APR1400 was the first non-US reactor model to receive the certification in the United States.

The Advanced Power Reactor 1400 MWe (APR1400) is a standard evolutionary Advanced Light Water Reactor (ALWR) developed in 2002. The design is based on the experience that has been accumulated through the development, construction, and operation of OPR1000, Optimum Power Reactor 1000MWe, the first standard Pressurized Water Reactor (PWR) plant in Korea. The APR1400 also utilized state-of-the-art proven technology and incorporated a number of advanced design features to meet the utility's needs for enhanced economic goals and to address the new licensing safety issues and requirements for an improved plant safety.

During the 1980s, Korea launched the technology self-reliance program on every aspect of a nuclear power plant construction project and accomplished the nuclear technology self-reliance through the OPR1000 design and construction projects. Currently, a total of 12 OPR1000 units are in operation with excellent performance and availability. The repeated construction and subsequent operation of OPR1000 units brought forth internationally competitive construction technology and outstanding plant operation and maintenance capabilities. Based on the self-reliant technology and experience accumulated through the design, construction, operation and maintenance of OPR1000, Korea launched the APR1400 development project in 1992 and completed its standard design in 2002.

The reactor type and plant design concept of APR1400 is the result of a two-year research program, Phase I, which was completed in 1994. During this phase, the advance PWR plant designs being developed worldwide were reviewed, and the major design goals for safety and performance as well as the design concept were established to meet the future demand for ALWRs. The safety and economic goals for APR1400 were set through a comparative study among the ALWR designs available worldwide and the utility requirements on ALWRs. The major design requirements for safety and performance goals set for APR1400 are listed in Table 1.

Table 1. APR1400 DESIGN REQUIREMENT FOR SAFETY AND PERFORMANCE GOALS

General Requirement	**Performance requirements and economic goals**
Type and capacity : PWR, 1400 MWe Plant lifetime : 60 years Seismic design : SSE 0.3g Safety goals : Core damage frequency < $1.0E^{-5}$/RY Containment failure frequency < $1.0E^{-6}$/RY Occup. radiation exposure < 1 man -Sv / RY	Plant availability : greater than 90% Unplanned trips : less than 0.8 per year Refueling interval : 18 months or longer Construction period : 48 months (N-th plant) Economic goal : 20% cost advantages over competitive energy sources

The basic design satisfying the above design requirements was developed during Phase II. Also, a series of experimental research on the advanced design features newly introduced in the design was performed in parallel with plant design development. The Nuclear Steam Supply System (NSSS) design has been completed with the general arrangement and the major component specifications. With a complete Balance Of Plant (BOP) design, the Standard Safety Analysis Report (SSAR) was developed. At the end of the basic design development in early 1999, a further design optimization was performed to improve the economic competitiveness, operability, and maintainability while maintaining the overall safety goal during Phase III. The APR1400 design optimization has been completed in 2001 and acquired the standard design approval from Korean regulatory body in May 2002. The APR1400 was determined to be built as the next nuclear power plant domestically following 12 units of OPR1000 being operated.

Two(2) APR1400 units have been built and now in commercial operation and 8 units are under construction. Shin-Kori units 3&4, which is the first APR1400 plant, has already received operation license and started commercial operation in December 2016 for unit 3 and August 2019 for unit 4, respectively. Shin-Hanul units 1&2, the second APR1400 plant, is in final construction stage with the goal of commercial operation in October 2020 for unit 1 and August 2021 for unit 2. Shin-Kori units 5&6, the third APR1400 plant, is under construction and in early stage. Barakah units 1, 2, 3&4, the first exported APR1400 plant, is under construction and the unit 1 has received operation license in February 2020.

3.1. Deployed Reactors

3.1.1. Republic of Korea

SKN 3&4 construction project, which is the first twin APR1400 unis, was launched by KHNP (Korea Hydro and Nuclear Power Company) with the major supply and construction contracts signed in August 2006. Site grading started in September 2007 and Construction Permit (CP) was issued by Korean government on April 15, 2008. The first concrete pouring for Unit 3 was done on October 15, 2008 as scheduled, while Unit 4 was done on August 19, 2009. After a successful structural and building construction as well as main equipment manufacturing, the Unit 3 reactor vessel was installed on July 15, 2010, and the Unit 4 reactor vessel was installed on July 18, 2011.

The Operating License (OL) for Unit 3 was issued on October 30, 2015, and on February 1, 2019 for Unit 4. Through successful commissioning and startup tests, commercial operation was started on December 20, 2016 for Unit 3, and August 29, 2019 for Unit 4. The site view of Shin-Kori units 3 and 4 is presented below.

Figure 15. Shin-Kori units 3&4 Site View

The second project for the construction of Shin-Hanul units 1&2(SHN 1&2) is in its final stages with the goal of commercial operation in October 2020 for Unit 1 and August 2021 for Unit 2. The third project for the construction of Shin-Kori units 5&6(SKN 5&6) is in progress with the target commercial operation in March 2023 for Unit 5 and in June 2024 for Unit 6.

The design of SHN 1&2 and SKN 5&6 is basically the same as SKN 3&4 except that the MMIS and the Reactor Coolant Pumps (RCP) are manufactured and supplied by domestic vendors. The MMIS and RCP are the results of years of R&D effort funded by government.

3.1.2. United Arab Emirates (UAE)

On December 27, 2009, the UAE and Korea signed a historical prime contract for supplying 4 units of APR1400 to the UAE, the first nuclear power plant in the Arab countries, as shown in Figure 16. Barakah located on the Arabian Gulf coast, west of Abu Dhabi was selected as the construction site. The Preliminary Safety Analysis Report (PSAR) of Barahah Nuclear Power Plant (BNPP) Units 1&2 was submitted to the Federal Authority Nuclear Regulation (FANR) in December 2010 and the Construction License (CL) was issued in July 2012. The construction of Unit 1 begun in July 2012 following the first of nuclear safety concrete, while Unit 2 begun in May 2013. The FSAR of BNPP Units 1&2 was submitted to the FANR in March 2015 and the unit 1 has received operation license in February 2020.

The BNPP design was based on its reference plant, SKN 3&4. However, a number of design changes have been implemented mainly due to the site specific environmental conditions that are different from the reference plant such as ambient seawater & air temperatures, sandstorm, salinity of seawater/groundwater, and 50 Hz of the grid frequency. In addition, major reinforced safety and security design addresses cyber security, physical protection, severe accident management and man-made and natural disasters such as aircraft crash and Fukushima accident.

Figure 16. Barakah Nuclear Power Plants

3.1.3. Compliance with worldwide design and licensing regulations

The APR1400 is internationally recognized for its safety and performance. The APR1400 has obtained design certification from U.S. Nuclear Regulatory Commission (NRC), and also won the certification from European Utility Requirements (EUR) organization. In 2017, EUR organization certified the APR1400 design for its compliance with the EUR. In 2019, the U.S. NRC issued final Design Certification (DC) for the APR1400 design. The acquisition of DC is significant in that APR1400 was the first non-US reactor model to receive the certification in the United States.

4. *Safety Concept*

SUMMARY FOR BOOKLET

Safety is a requirement of paramount importance for nuclear power. One of the APR1400 development policies is to increase the level of safety significantly. Safety and economics in nuclear power plants are not counteracting each other but can move in the same direction, since the enhancement of safety will also yield an improved protection of the owner's investment. Therefore, safety has been given top priority in developing the new design. To implement this policy, in addition to the plant being designed in accordance with the established licensing design basis to meet the licensing rules, APR1400 was designed with sufficient safety margin in order to improve the protection of the investment, as well as the protection of the public health.

APR1400 is designed with the concept of defence-in-depth which is implemented to five levels of protection, including successive barriers preventing the release of radioactive material to the environment. The defence-in-depth strategy is to prevent accidents and, if failed, to limit their potential consequences and prevent any evolution to more serious conditions.

The consequences arising from all anticipated operational occurrences and accident conditions have to be addressed in the safety analysis. This includes accidents that have been taken into account in the design (referred to as design basis accidents, DBAs) as well as beyond design basis accidents including severe accidents for facilities and activities where the radiation risks are high. Safety analysis of APR1400 has been performed to demonstrate the performance of the components, its operating systems, its safety systems to cope with a wide spectrum of anticipated operational occurrences and postulated accidents. The safety analysis is based on deterministic methods complemented by probabilistic safety assessment (PSA). The deterministic safety analyses demonstrate that the safety functions are accomplished and adequate protection for the environment is provided without any violation of the acceptance criteria for those events categorized in design basis accident.

4.1. Safety Philosophy and Implementation

4.1.1. Safety concept, design philosophy and licensing approach

The nuclear industry which differs from other industrial sectors always puts highest priority to nuclear safety. For the purposes of this publication, 'safety' means the protection of people and the environment against radiation risks, and the safety of facilities and activities that give rise to radiation risks. One of the APR1400 development policies is to increase the level of safety significantly. Safety and economics in nuclear power plants are not counteracting each other but can move in the same direction, since the enhancement of safety will also yield an improved protection of the owner's investment. Therefore, safety has been given top priority in developing the new design.

The main design philosophy of the APR1400 is the enhancement of safety using proven technologies and significant experiences gained in design, construction, and operation of nuclear power plants in Korea. APR1400 was designed with sufficient safety margin in order to improve the protection of the investment, as well as the protection of the public health.

Quantitative safety goals for the design were established in a probabilistic approach:

- The total core damage frequency (CDF) shall not exceed 10^{-5}/RY for both internal and external initiating events and 10^{-6}/RY for a single event and an incident occurred in the high pressure condition.
- The containment failure frequency shall be less than 10^{-6}/RY
- The whole body dose at the site boundary shall not exceed 0.01 Sv (1 rem) for 24 hours after initiation of core damage with a containment failure.

4.1.2. Operational safety margins, at initial load, end-of-life, and during load following

The safety margin of operating reactors is defined as the difference or ratio in physical units between the limiting value of an assigned parameter the surpassing of which leads to the failure of a system or component, and the actual value of that parameter in the plant. The existence of such margins assure that nuclear power plants (NPPs) operate safely in all modes of operation and at all times. The most important safety margins are related with physical barriers against release of radioactive material, such as fuel matrix and fuel cladding RCS pressure and surrounding public dose.

4.1.3. Defence-in-depth description

The defence-in-depth (DiD) concept in the APR1400 design is implemented to five levels of protection, including successive barriers preventing the release of radioactive material to the environment. The DiD strategy is to prevent accidents and, if failed, to limit their potential consequences and prevent any evolution to more serious conditions. DiD is generally structured in five levels. Five levels of protection are implemented in such a way that when one level fail it makes the subsequent level come into play as below:

- Level 1 : Prevention of abnormal operation and failure by means of conservative design and high quality in construction and operation
- Level 2 : Detect and control of abnormal operation and failures by means of control and limiting systems and other surveillance features
- Level 3: Limit radiological release and prevent deterioration to core melt conditions by engineered safeguards and emergency procedures against postulated events as follows:
  - Level 3a : Control of postulated single initiating events and by means of engineered safety features (ESF)
  - Level 3b : Control of postulated complex sequences and multiple failures by means of ESF or diverse safety features (DSF)
- Level 4 : Control of severe accidents to limit off-site release by means of severe accident mitigation system and severe accident management
- Level 5 : Mitigation of radiological consequences of significant releases of radioactive material by means of off-site emergency response

4.2. Transient/Accident Behaviour

4.2.1. Description of limiting DBAs and severe accidents

A set of design basis accidents (DBAs) is postulated and accommodated by the design of the APR1400. The consequences of the DBAs fall within the acceptance limits established to protect safety of the public and the plant staff, as well as to provide an appropriate level of plant protection. The DBAs of APR1400 are consistent with the postulated initiating events described in safety review guidelines for light water reactors categorized as below:

- Increase in heat removal by the secondary system
- Decrease in heat removal by the secondary system
- Decrease in reactor coolant system flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory
- Radioactive release from a subsystem or component

A limiting list of DBAs for APR1400 is as follows:

- Steam system piping failures inside and outside containment
- Feedwater system pipe breaks
- Single reactor coolant pump rotor seizure
- Reactor coolant pump shaft break
- Single control element assembly withdrawal
- Control element assembly ejection
- Steam generator tube rupture
- Loss of coolant accident

The acceptance criteria for the DBAs are defined in safety review guidelines for light water reactors. The acceptance criteria for anticipated operational occurrence (AOO) are based on maintaining the reactor coolant system pressure boundary integrity and ensuring that specified acceptance fuel design limits are not exceeded, and offsite radiological consequences are acceptable. The typical acceptance criterion for AOO is that the maximum reactor coolant system pressure shall be less than 110% of design pressure.

In the APR1400 design, severe accidents are addressed as follows:

- For phenomena likely to cause early containment failure, for instance, within 24 hours after accidents, mitigation systems shall be provided or design should address the phenomena although the probability for such accidents is low
- For phenomena which potentially lead to late containment failure if not properly mitigated, the mitigation system or design measures should be considered in conjunction with the probabilistic safety goal and cost for incorporating such features to address the phenomena

This approach is to enhance the effectiveness of investment on safety by avoiding undue over-investment on highly improbable accidents. Also, a realistic assessment is recommended for severe accident analyses.

4.2.2. Safety systems to cope with design basis accidents (DBA)

The plant protection system (PPS) is a safety system that includes electrical, electronic, network, mechanical devices, and circuits and performs the protective functions. The reactor protection system (RPS) is the portion of the PPS that acts to trip the reactor when required. The engineered safety features actuation system (ESFAS) is the portion of the PPS that activates the engineered safety features (ESF) systems to cope with DBAs for mitigation. The ESF systems are described as follows.

Safety injection system

The safety injection system (SIS) is designed to provide core cooling in the unlikely event of a loss of coolant accident (LOCA). The SIS limits fuel damage to maintain a coolable core geometry, limits the cladding metal-water reaction, removes the energy generated in the core and maintains the core subcritical during the extended period of time following a LOCA.

The SIS accomplishes these functional requirements using redundant active and passive injection subsystems. The SIS is comprised of four independent mechanical trains without any tie line among the injection paths as shown in Figure 17. The active portion of the SIS consists of four mechanically separated trains, each consisting of a safety injection (SI) pump and associated valves. Each SI pump is provided with its own suction line from the incontainment refueling water storage tank (IRWST), and its own discharge line to a direct vessel injection (DVI) nozzle on the reactor vessel. The passive portion consists of four identical pressurized safety injection tanks (SITs) equipped with a Fluidic Device (FD). To satisfy the LOCA performance requirements, each train provides 50% of the minimum injection flow rate for breaks larger than the size of a direct vessel injection line. For breaks equal to or smaller than the size of a direct vessel injection line, each train has 100% of the required capacity.

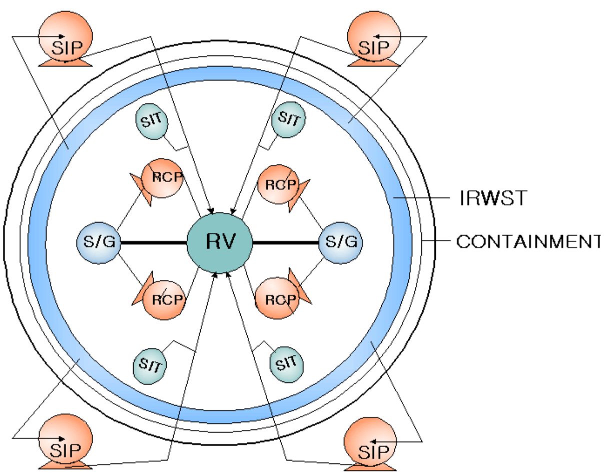


Figure 17. Safety Injection System

The SIS is capable of injecting borated water into the reactor vessel to mitigate accidents other than LOCAs. Safety injection would be initiated in the event of a steam generator tube rupture, steam line break or a CEA ejection incident. The borated water injected by the SIS provides inventory and reactivity control for these events.

Shutdown cooling system

The shutdown cooling system (SCS) is a safety-related system that is used to reduce the temperature of the reactor coolant system (RCS) in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. The initial phase of a cooldown is accomplished by heat rejection from the steam generators (SGs) to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 176.7°C (350°F) and 31.6 kg/cm^{2}A (450 psia), the SCS is put into operation for normal shutdown cooling to reduce the RCS temperature to the refueling temperature, and to maintain this temperature during refueling.

Additionally, the SCS is used in conjunction with the atmospheric dump valves (ADVs) and the auxiliary feedwater system to cooldown the RCS following a small break loss of coolant accident (LOCA) (Refer to Section 6.3). The SCS is also used subsequent to steam and feedwater line breaks, steam generator tube ruptures, and is used during plant startup prior to RCP restart to maintain flow through the core. After an accident the SCS can be put into operation when the RCS pressure and temperature are below approximately 193.3°C (380°F) and 28.1 kg/cm2A (400 psia).

Containment spray system

The containment spray system (CSS) is a safety grade system designed to reduce containment pressure and temperature from a main steam line break or loss of coolant accident and to remove fission products from the containment atmosphere following a loss of coolant accident. Fission product removal is required so that activity at the site boundary due to radioactive release is reduced. No spray additive is required for pH control during the initial stage of a LOCA. However, additive is required for post-accident pH control of the spray water.

Auxiliary feedwater system

The auxiliary feedwater system (AFWS) provides an independent safety-related means of supplying feedwater to the steam generator(s) secondary side for removal of heat and prevention of reactor core uncovery during emergency phases of plant operation. The auxiliary feedwater system is an independent 2-division system, one for each SG, and each division has 2 trains. The reliability of the AFWS has been increased by use of one 100% capacity motor-driven pump and one 100% capacity turbine-driven pump for diversity and redundancy and one dedicated safety-related auxiliary feedwater storage tank in each division as a water source in addition to the non-safety grade condensate storage tank as a backup source.

The AFWS is designed to be automatically or manually initiated, supplying feedwater to the steam generators for any event that results in the loss of normal feedwater and requires heat removal through the steam generators, including the loss of normal onsite and normal offsite AC power.

Following the event, the AFWS maintains adequate feedwater inventory in the steam generator(s) for residual heat removal and it is capable of maintaining hot standby and facilitating a plant cooldown (at the maximum administratively controlled rate of 41.7 °C/hr (75 °F/hr)) from hot standby to shutdown cooling system initiation.

Safety depressurization and vent system

The safety depressurization and vent system (SDVS) is a dedicated safety system designed to provide a safety grade means to depressurize the RCS in the event that pressurizer spray is unavailable during plant cooldown to cold shutdown and to depressurize rapidly the RCS to initiate the feed and bleed method of plant cooldown subsequent to the total loss of feed-water event. The pilot operated safety relief valves (POSRVs) are employed for feed and bleed operation. This system establishes a flow path from the pressurizer steam space to the IRWST.

Main steam system

Steam is generated in two steam generators by heat transferred from the reactor coolant system to the feedwater. Five ASME Code spring loaded main steam safety valves (MSSVs) are provided for each individual main steam line for protection against overpressurization of the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve.

A main steam atmospheric dump valve (MSADV) is provided on each main steam line upstream of the safety valves and the MSIVs. Each main steam line is provided with a main steam isolation valve for positive isolation against forward steam flow and isolation against reverse flow. Each MSIV is provided with a bypass around it for warm-up of the steam lines downstream of the isolation valves and pressure equalization prior to admitting steam to the turbine.

4.2.3. Safety systems to cope with severe accidents (beyond design basis accidents)

The facilities to mitigate severe accidents are designed to meet the procedural requirements and criteria of the U.S.NRC regulations, including the Three Mile Island (TMI) requirements for new plants as reflected in 10 CFR 50.34 (f) and SECY-93-087. The severe accident management systems consists of a large dry pre-stressed concrete containment, Hydrogen Mitigation System (HMS), Cavity Flooding System (CFS), Ex-Reactor Vessel Cooling System (ERVCS), Safety Depressurization & Vent System (SDVS), Emergency Containment Spray Backup System (ECSBS).

Containment building

In order to maintain the integrity of the RCB and prevent the leakage of radioactive materials against severe accidents, the RCB is designed to have enough free volume for the load to be below ASME Section III Service Level C in 24 hours after severe accidents and to keep hydrogen concentration under 13% in case of 75% oxidation of fuel clad-steam and the 1/4 inch steel plate is installed in the inside of the RCB. In addition, the RCB is constructed with the concrete having compressive strength of 6,000psi after 91 days curing.

Hydrogen mitigation system

During degraded core accident, hydrogen will be generated at a greater rate than during the design basis LOCA. The HMS is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in containment to 10% for degraded core accidents. The HMS consists of a system of Passive Auto-catalytic Recombiners (PARs) complemented by glow plug igniters installed within the containment. The PARs serve for accident sequences in which mild or slow release rates of hydrogen are expected, and are installed uniformly in the containment. Whereas, the igniters supplement PARs under the accident of very low probability in which very rapid release rates of hydrogen are expected, and are placed near source locations to facilitate the combustion of hydrogen in a controlled manner such that containment integrity is maintained.

Cavity flooding system

The APT1400 reactor cavity adopts a core debris chamber, which is designed to have the heat transfer area of corium more than 0.02m2/MWt. The flow path of reactor cavity is designed to be convoluted to hinder the transfer of core debris to the upper containment. This design prevents Direct Containment Heating (DCH) by core debris.

The CFS consists of two trains connected with IRWST and two isolation valves are installed on each line as shown in Figure 18. When two isolation valves are open during severe accidents, the cavity cooling water is supplied from the IRWST to the reactor cavity driven by the gravity induced from the difference of water head in-between and then cools down the core debris in the reactor cavity, scrubs fission product releases, and mitigates the molten corium concrete interaction (MCCI).

External reactor vessel cooling system

The ERVCS is implemented as a severe accident mitigation system used for the purpose of in-vessel retention of corium under hypothetical core-melting severe accident conditions. The ERVCS shall be used only under the severe accident condition and thus is designed on safety margin basis. As shown in Figure 18 one train of shutdown cooling pump, with related valves, pipes, and instrumentation & controls, is provided for initial reactor cavity flooding to the level of hot leg. After the initial flooding by the shutdown cooling pump, the Boric Acid Makeup Pump (BAMP) is utilized to refill the reactor cavity, at a flow rate greater than that of boiling caused by decay heat from the molten core.

The ERVCS is designed to be manually operable only when the core exit temperature reaches a certain temperature following a severe accident. The operating procedure for the ERVCS was developed through severe accident analysis and probabilistic safety assessment. The gravity driven Cavity Flooding System (CFS) provides flooding of the reactor cavity below the reactor vessel. The CFS is a backup system used in case that ERVCS is unavailable and provides corium cooling, should the reactor vessel fail.

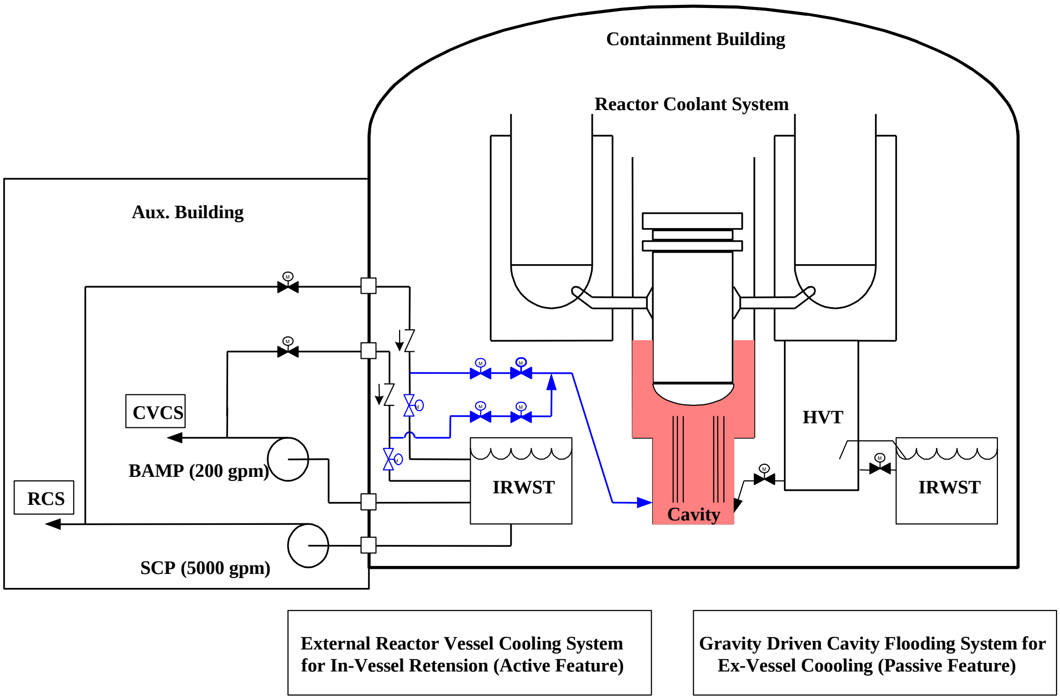


Figure 18 External reactor vessel cooling system and cavity flooding system

5. *Fuel and Fuel Cycle*

5.1.Fuel Cycle and Fuel Options

APR1400 is a typical PWR plant using slightly enriched uranium and, hence, is not designed as a breeder or a high-converter reactor. However, the reactor core and other related systems are designed to use MOX fuel up to 1/3 of core. The spent fuel treatment plan is beyond the plant design scope.

The core is designed for an operating cycle of above 18 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of more than 10% to enhance safety and operational performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber of gadolinium (Gd_2O_3) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by means of 61 fixed In-Core Instrumentation (ICI) assemblies.

Loading mixed oxide (MOX) fuel up to 1/3 core is considered in the core design. Eight additional reserve CEAs is installed to increase the reactivity control capability, if necessary, for MOX fuel loadings. Also, the APR1400 core is designed to be capable of daily load following operation.

The HIPER16™ fuel design has the capability of a batch average discharge burn-up higher than 55,000 MWD/MTU and the HIPER16™ design has increased overpower margin in comparison with the previous fuel design (PLUS7™). The HIPER16™ mid-grid design has high through-grid dynamic buckling strength for the enhanced seismic performance. The top nozzle has easy reconstitutability features and holddown spring force has optimized to reduce the fuel assembly bow. The guide tube has high seismic load capability and inner dashpot tube to minimize the fuel assembly bow. The debris filtering and capturing features are implemented in the bottom grid by combining the debris filtering bottom grid and the long bottom end plug to reach the target of zero fuel failure. The bottom nozzle has a low pressure drop features with rectangular flow holes.

The integrity of HIPER16™ fuel has been enhanced by increasing the fretting wear resistance and debris filtering efficiency. The optimized holddown spring force will reduce the HIPER16™ fuel assembly bow. The safety of HIPER16™ fuel has been enhanced by increasing the seismic performance which is related to the spacer grid crush strength and dynamic stiffness.

5.2. Spent Fuel and Waste management

Radioactive waste management systems include systems, which deal with liquid, gaseous and solid waste, which may contain radioactive material. The Liquid, Gaseous and Solid Waste Management Systems are located in the compound building.

5.2.1. Liquid Waste Management System (LWMS)

The design objectives of the LWMS is to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor radioactive liquid waste. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The processed liquid radioactive waste is sampled prior to release from monitor tanks and radiation monitors are provided in the discharge line to provide for a controlled monitored release.

The LWMS provides a means to collect, store, process, sample, and monitor radioactive liquid waste. The LWMS consists of collection tanks, process filter, Advanced Liquid Treatment System (ALTS) packages, process pumps and vessels, monitor tanks, and appropriate instruments and controls to permit most operation to be conducted remotely. Radioactive wastes are segregated by routing to an initial collection sump or tank. This permits more effective processing of each type of waste and may lead to reduced solid waste volumes. Segregation is accomplished by means of a subsystem that processes different waste categories. The LWMS is divided into the following major subsystems:

- Floor Drain Subsystem
- Equipment Waste Subsystem
- Chemical Waste Subsystem
- Radioactive Laundry Subsystem (RLS)

5.2.2. Gaseous Waste Monitoring System (GWMS)

The GWMS is provided with radiation monitors that monitor the discharge from the charcoal delay beds upstream of the discharge to the Compound Building HVAC System. The GWMS consists of two subsystems; the process gas subsystem and the process vent subsystem. The system is designed to collect, process, and monitor releases from all system input streams.

<u>Process Gas Portion of the GWMS</u>

Process gases contain radioactive xenon and krypton fission products from fuel and tramp uranium on fuel surfaces. The process gas portion of the GWMS receives fission gases from the Process Gas Header (PGH) and uses charcoal delay beds to delay the process gases for decay prior to release. The primary input source to the PGH is gases stripped from reactor coolant by the CVCS Gas Stripper and Volume Control Tank. The Reactor Drain Tank process vent surge volumes comprise the balance of the hydrogenated gaseous inputs to the PGH. The flow in the PGH primarily consists of hydrogen and noble gases with some trace quantities of other fission gases and water vapor. The removal of fission gases by the gas stripper maintains the fission gas concentration in the reactor coolant at a low residual level. This minimizes the escape of radioactive gases during maintenance on the reactor coolant system and releases resulting from leakage of reactor coolant.

<u>Process Vent Portion of the GWMS</u>

The process vent portion of the GWMS is designed to collect the low activity aerated gas streams from the potentially contaminated vents headers in the Containment, Auxiliary Building, Turbine Building and Compound Building. The process vents, except from the condenser evacuation system, are monitored, filtered (if required) through the building HVAC systems, and released through the unit vent. The condenser evacuation system is monitored and then discharged through the unit vent.

5.2.3. Solid Waste Management System (SWMS)

The SWMS is designed to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor solid waste. The SWMS processes both wet solid active waste and dry active waste for shipment to a licensed disposal site.

Primary functions of the SWMS include providing means by which spent resin, filters, etc. from the LWMS and primary letdown systems are processed to ensure economical packaging within regulatory guidelines, as well as handling dry, low activity wastes for shipment to a licensed disposal facility. The SWMS is subdivided into the following subsystems:

- Spent resin transfer and storage system
- Filter handling system
- Dry active waste sorting and segregation system
- Spent resin drying system (SRDS)

5.3. Fuel handling and transfer systems

The fuel handling system is designed for a safe and rapid handling and storage of fuel assemblies from the receipt of fresh fuel to the shipment of spent fuel.

The major equipment of the system comprises the refuelling machine, the CEA change platform, the fuel transfer system, the fresh fuel elevator, the CEA elevator and the spent fuel handling machine. The refuelling machine is located in the containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The spent fuel handling machine, located in the fuel building, carries fuel to and from the fuel transfer system, the fresh fuel elevator, the spent fuel storage racks, and the spent fuel shipping cask areas.

6. *Safeguards and Physical Security*

The APR1400 design and plant layout considered safety and security in various ways in which the design and configuration inherently protects the plant against human induced malevolent external impacts and insider action. The physical security system is designed in accordance with the applicable regulations and is expected to provide protection against malevolent acts of sabotage with high assurance.

The main design features for APR1400 safety and security are as follows:

- Thick concrete walls for exterior and a large number of interior walls protect those equipments important to safety and provide a significant deterrent to penetration. The Auxiliary Building is physically separated in 4 quadrants, which provides adequate physical separation and barrier.
- The entry control point to the plant is centralized with security facilities and located in the compound building only for both units for the twin unit plants.
- A robust vehicle barrier system that is located at a safe standoff distance.
- Fencing is employed to establish a perimeter boundary at a sufficient distance such that under normal circumstances, security response force personnel are able to identify and engage a potential land based assault.
- An intrusion detection system is employed adjacent to the protected area boundary fencing to provide indication of unauthorized attempts to enter the protected area.
- A closed circuit television network is used to provide remote monitoring of the protected area boundary.

An access control system is utilized to permit only properly authorized personnel into designated areas of the facility